Joint DOE-EPRI Strategic Research and Development Plan to Optimize U.S. Nuclear Power Plants

VOLUME II

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Table of Contents

Generation Optimization Projects 3-56

1.0 Introduction

The *Joint DOE-EPRI Strategic Research and Development Plan to Optimize U.S. Nuclear Power Plants* is a market-based assessment of energy supply R&D needs for current plants. It is the strategic planning document for industry and government collaboration on nuclear energy R&D needs, and has become the foundation and planning document for the jointly funded Nuclear Energy Plant Optimization (NEPO) Program.

The NEPO Program develops key technologies to help ensure that our nation's existing nuclear power plants can continue to deliver reliable and affordable energy supplies up to and beyond their initial 40-year license period. The program works to resolve open issues related to plant aging, and applies new technologies to improve plant reliability, availability, and productivity. The research conducted under the NEPO program addresses the long-term effects of component aging; improved nuclear plant capacity factors; optimization through efficiency and productivity improvements; and increased power output while maintaining high levels of safety. The NEPO program is conducted by DOE in cooperation with EPRI and the nuclear industry with joint management and cost sharing.

Volume I of the Joint Plan addresses the overall plan, goals and needs assessment for operating nuclear power plants in the United States. Volume II identifies the R&D tasks being proposed for the coming year. The Joint Plan was first published in March 1998 and both volumes of the plan were revised in October 2000 to support joint DOE-EPRI planning for fiscal year 2001. This revision updates Volume II only and contains high priority projects proposed for fiscal year 2002. DOE and EPRI worked with electric utilities, national laboratories, the Nuclear Regulatory Commission, and other stakeholders to perform an in-depth analysis of the issues facing commercial nuclear power plants, research needed to resolve these issues, and research being conducted domestically and internationally to arrive at the research projects identified in this update.

2.0 Funding Summary

The following table provides a summary of DOE funding requirements for all the projects included in this volume. EPRI's cost-share is not included in this Summary.

Project No.	Project Title	FY00 to FY01 Actual DOE ($K)	FY02 Proposed DOE ($K)	FY03 and Beyond Proposed DOE ($K)	Total DOE ($K)
R&D Area: Non-destructive Examination					
3-1	Steam Generator Non-Destructive Examination Test Mockup Facility and Tube Degradation Database	750	260	0	1,010
3-2	Advanced Eddy-current Inspection System for Detection and Characterization of Defects in Steam Generator Tubes	552	290	0	842
3-218	Advanced Millimeter-Wave Sensor for Non-Contact Non-Destructive Evaluation of Cast Stainless Steel Pipes	0	250	250	500
3-228	Microwave Sensor for Non-Contact Condition Monitoring of Containment	0	250	750	1,000
3-229	Non-Destructive Inspection of Inaccessible Portions of NPP Metallic Pressure Boundaries	0	300	900	1,200
R&D Area: Electrical Cables and Equipment					
3-7	Develop Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable	905	330	0	1,235
3-8	Develop Condition Monitoring (CM) Techniques and Database for Electrical Cable	1,196	335	0	1,531
3-29	Motor Rewind Insulation System Development and Qualification for Harsh Environments	262	390	281	933

Project No.	Project Title	FY00 to FY01 Actual DOE ($K)	FY02 Proposed DOE ($K)	FY03 and Beyond Proposed DOE ($K)	Total DOE ($K)
	R&D Area: Mechanical Properties and Aging				
3-13	Mechanical Behavior of Irradiated Structural Stainless Steels	807	465	3,813	5,085
3-24	Fatigue	1,160	660	500	2,320
3-27	Assessment of Aging Effects on Components and Structures from Nuclear Power Plants	400	500	1,092	1,992
3-30	Irradiation Induced Swelling and Irradiation Enhanced Stress Relaxation of PWR Reactor Core Internal Components	1,054	400	396	1,850
3-211	Thermal Aging Embrittlement of PWR Metals	0	250	250	500
3-212	Aging Data for Long-term Reliability of Systems, Structures, and Components (SSCs)	0	175	350	525
3-221	Synergistic Effects of Irradiation and Thermal Aging in Cast Stainless Steels	0	565	400	965
3-223	Revision to ASME Section XI Appendix G, RPV Pressure – Temperature Limits	0	400	350	750
3-224	Master Curve Fracture Toughness Implementation	0	300	950	1,250
	R&D Area: Corrosion and Stress Corrosion Cracking				
3-207	Mitigation of Initiation and Growth of PWSCC in Alloy 600 and 82/182 Weld Metals	0	125	450	575
3-209	Validation of BWR Fluence Models and Weldability of Internals	0	500	0	500
3-210	Low Temperature Hydrogen Cracking of Ni-base Alloys and Weld Metals	0	75	75	150

Project No.	Project Title	FY00 to FY01 Actual DOE ($K)	FY02 Proposed DOE ($K)	FY03 and Beyond Proposed DOE ($K)	Total DOE ($K)
3-222	Develop a Predictive Model for Pitting Corrosion of Heat Exchanger Tubes	0	175	150	325
3-225	Crack Propagation Study of PWR Core Internals using Small Specimen Designs and Test Techniques	0	400	1,400	1,800
3-226	Developing Ni-Alloy Welds Resistant to Stress Corrosion Cracking	0	300	500	800
R&D Area: Instrumentation and Control					
5-110	Guidelines for Hybrid Control Rooms	200	370	435	1,005
5-113	On-line Monitoring of Non-Redundant Sensors for Improved Performance	109	260	141	510
5-201	LOCA Qualifiable Digital Transmitter Based on Fiber Optics	0	250	150	400
5-202	Guidelines for the Monitoring of Aging of Nuclear Power Plant I&C System Electronic Boards	0	85	110	195
5-204	Qualification of Commercial Digital Components for Replacement of Obsolete Equipment in Nuclear Safety Systems	0	341	1,925	2,266
5-206	Development of a Safety-Critical Architecture for Embedded Applications Using Commercial-Off-The Shelf (COTS) Hardware and Software	0	125	337	462
5-213	Using Deterministic Analysis Tools and Statistic Combinations of Uncertainties (SCU) Methodology to Support Instrument Calibration	0	200	300	500
5-220	Guidelines for Wireless Technologies in Nuclear Power Plants	0	75	75	150

Project No.	Project Title	FY00 to FY01 Actual DOE ($K)	FY02 Proposed DOE ($K)	FY03 and Beyond Proposed DOE ($K)	Total DOE ($K)
5-221	Development of New Algorithms for use in Digital Protection and Monitoring Systems	0	250	1,000	1,250
5-227	Improved Temperature Measurements at US Nuclear Power Plants	0	200	175	375
R&D Area: Risk Technologies					
5-106	Low Power and Shutdown Probabilistic Risk Assessment (PRA) Qualitative Research Pilot Plant Proof of Concept / Low Power and Shutdown Human Reliability	284	270	31	585
5-117	R&D Needs to Address Potential Nuclear Plant Vulnerabilities Arising From Transmission Grid Voltage Inadequacies	152	150	138	440
5-214	LOCA Initiating Event Frequency Derivation	0	150	0	150
5-218	Nuclear Plant Safety Risk Management	0	200	275	475
R&D Area: Nuclear Fuel & Coolant Chemistry					
5-103	Dry Storage of Spent Fuel with Burnup in Excess of 45 GWd/MTU	0	150	0	150
5-222	Developing An Optimized Procedure for Preparation of Spent Fuel Prior to Dry Storage	0	300	600	900
5-232	Nano-scale Microstructural Analysis of PWR and BWR Cladding at Burnups exceeding 45 GWd/MTU: Correlation to Physical Properties	0	250	800	1,050
5-233	Cladding Oxide Spallation Mechanisms	0	500	500	1,000

Project No.	Project Title	FY00 to FY01 Actual DOE ($K)	FY02 Proposed DOE ($K)	FY03 and Beyond Proposed DOE ($K)	Total DOE ($K)
5-234	Feasibility Determination For the Use of Enriched Boric Acid to Avoid PWR Axial Offset Anomaly (AOA)	0	341	750	1,091
5-235	Dissolution of Fuel Cladding Oxide in High Duty PWR Cores	0	115	60	175
5-236	Optimizing Coolant Chemistry in BWRs Using Depleted Zinc Oxide and Noble Metal Chemical Application (NMCA)	0	200	50	250
5-237	Root Cause Investigation of Fuel Rod Oxide Spallation and Noble Metal Chemical Application (NMCA)	0	500	500	1,000
	Totals	7,831	12,977	21,209	42,017

3.0 Project Descriptions

Aging Management Projects

3-1 Steam Generator Non-Destructive Examination (NDE) Test Mockup Facility and Tube Degradation Database

Principle Objective: The objective of this work is to provide the capability to assess the effectiveness of NDE techniques, current and emerging, for the detection and characterization of service induced cracks in steam generator tubes. Such studies will be used to benchmark and confirm evaluations based on laboratory grown cracks and other simulations of service induced flaws. The steam generator mock-up at Argonne developed in an USNRC program which has a wide variety of laboratory grown stress corrosion cracks (SCC), other laboratory grown SCC samples, and samples with service induced SCC (primarily SCC from the McGuire plant acquired through a USNRC program), will form the sample base for this effort. Maintaining interactions with industry will be an important aspect of this program.

Need: Steam generator degradation due to corrosion cracking of steam generator tubing is an extremely expensive problem for existing PWRs. Testing pulled tubes taken from operating reactors have shown that in many cases the cracking that occurs often does not compromise the structural integrity of the tube or permit significant leakage under normal operating or accident conditions. However, because the capability of NDE to detect defects has progressed much more rapidly than the capability to size or characterize defects, utilities have been forced to adopt a "plug on detection" policy to deal with crack indications. The development and qualification of improved techniques is dependent on the availability of suitable test specimens. Laboratory techniques to produce reasonably realistic cracks have been developed, but it desirable to be able to benchmark those results against results from actual stress corrosion cracks from operating reactors. Improvements in establishing integrity of tubes could save hundreds of thousands of dollars per outage.

Expected Duration: 4 years (FY 2000 – FY 2003)

Scope of Work:

<u>FY2002 Scope of Work</u>:

Work to be carried out in FY2002 is an extension of work begun in FY2000.

Metallographic analysis of field degraded test sections (primarily USNRC provided McGuire tube sections) will be carried out and compared to NDE results. The capability to assess the effectiveness of NDE techniques (including eddy current arrays and ultrasonic systems) for the detection and characterization of service induced cracks in steam generator tubes will be evaluated.

Work Scope for Subsequent Periods:

Additional efforts will be to extend the NDE effort to emerging NDE technology and to expand the database using additional field induced flaws.

FY02 Deliverables:

- Laboratory grown stress corrosion cracks for NDE evaluations.
- Reports on NDE capability for detection of field degraded tubing and assessment of eddy current arrays and ultrasonic systems.

FY02 Estimated Cost: $260 K

Total Estimated Cost: $1,010 K

3-2 Advanced Eddy-Current Inspection System for Detection and Characterization of Defects in Steam Generator Tubes

Principal Objective: The objective is to develop and test an advanced eddy current inspection system to improve the detection and characterization of defects in steam generator tubes. The focus of the work is to develop software algorithms that will provide a means to quickly, accurately and consistently detect and characterize steam generator tubing degradation from data acquired from eddy current array probes.

Need: The inspection of PWR steam generator tubing is an essential element to ensure safe and reliable operation of steam generators. The nuclear industry has made significant advancements in recent years with the development of probes to detect degradation in various regions of the steam generator tubing (e.g., expansion transitions, dents, u-bends, etc.). Unfortunately, multiple NDE techniques have been required to achieve optimum degradation detection capabilities over the entire tube length. This has led to increased inspection time, particularly when the slow rotating probe technology is the primary means to inspect specific regions on the tubing (e.g., expansion transitions) or flaw orientations (e.g., circumferential cracks). Recent development of array probe designs, such as the X-probe, offers the potential to quickly detect steam generator tube degradation in all regions of the tubing with only one probe. The challenges that this probe design creates include 1) the development of a means to quickly and consistently analyze the large amounts of data that are acquired and 2) the ability to characterize the degradation as precisely as possible to minimize the need to retest the location of "possible" degradation with other probe designs. The current limitation of the probes is that the quantity of the data acquired can not be efficiently analyzed in a manual mode. The development of a method to automate the analysis of array probe data will allow utilities to maximize the scheduler benefits of the probe, obtain consistent and repeatable results and reduce the dependency and costs associated with large teams of data analysts.

Expected Duration: 3 years (FY 2000 – FY 2003)

Scope of Work:

<u>FY2002 Scope of Work:</u>

FY 2002 is Year 3 of the project. In Year 3, the work scope will include effort on algorithms for tube burst pressure prediction. Other tasks are the finalization of the automated data analysis package design and the software implementation for the complete data analysis package. Proof-of-performance testing is carried out and project documentation is produced.
Task 1 (NEPO Project 3-1 Interface): Continue to integrate with NEPO Project 3-1. Where possible, obtain from NEPO Project 3-1 tube burst pressure data. The data will cover laboratory produced flawed tube samples and, if available, steam generator tubes with in-service cracks. If the data is unavailable, and where needed, complementary work to obtain the data will be performed in this task.

Task 2 (Burst Pressure Algorithm): Algorithms for prediction of the tube burst pressure based on the eddy current response for the degradation will be developed. The development work will use data from flawed tube samples. After the algorithms have been developed, testing of the tube burst pressure algorithms will be performed by comparing burst pressure predictions produced by the algorithms against burst testing data.

Task 3 (Software Package Implementation): Software implementation will be performed of the degradation characterization and burst pressure prediction algorithms. Specific requirements for platforms, operating systems, languages/toolkits will be met. The package will be completed. GUIs will be implemented and interfaced. Software will be produced as both (1) a stand-along package and (2) as separate modules that can be linked to commercially available data analysis software. The package will be compatible with the use of the X-probe (EC array).

Task 4 (Automatic Analysis Testing): Using the software package produced in Task 3, testing of the automatic analysis algorithm will be performed. The input will be signal data from array coils and the output will be tube burst pressure predictions. The test matrix will include flawed tube samples and, if available, in-service cracks from steam generator tubes. The accuracy of the algorithms to predict burst pressure will be evaluated. Also included will be the probability of flaw detection and the rate of false calls.

Task 5 (Project Documentation): Document the work in a two-volume final project report. Volume 1 will document the algorithms developed in each of the Tasks. Volume 2 will document the results on the testing of the accuracy of the integrated automatic data analysis algorithms. A software user's manual will also be produced.

FY2002 Deliverables:

- The tube burst pressure prediction algorithm will be produced.
- An automated eddy-current data analysis software package, ANTARES, will be completed to include the burst pressure prediction algorithm and the tube degradation characterization algorithm. The package will be performance tested.
- A project final report describing algorithms, benchmark data, and assessment of accuracy of automatic data analysis will be produced. A software user's manual will also be produced.

FY02 Estimated Cost: $290K

Total Estimated Cost: $842K

3-7 Develop Empirical Data to Characterize Aging Degradation of Polymers Used in Electrical Cable

Principle Objective: Quantifying and understanding aging of important nuclear power plant electrical cable materials by 1) developing correlations between elongation and ultrasensitive oxygen consumption (UOC) data in order to quantitatively test Arrhenius extrapolation model, 2) deriving an understanding of the basis of inverse temperature/Lazarus recovery phenomena and 3) developing and testing the Wear-out approach for predicting residual material lifetimes.

Need: This project will lead to more confident predictions of cable lifetimes as well as a method for estimating residual lifetimes of naturally aged cables that will serve as a secondary confirmation of the extrapolated predictions. The project will therefore minimize the necessity of expensive cable replacement.

Scope of Work:

FY2002 Scope of Work:

Subtask 1. Testing Arrhenius Using Ultrasensitive O_2 Consumption (on-going in FY2002).
Uncertainties related to long-term cable system operability and Arrhenius methodology calculations of qualified life are a significant regulatory concern. Older approaches use short-term, high temperature aging and elongation at break testing to derive Arrhenius activation energy (E_a) and then extrapolate to service conditions assuming E_a is unchanged. This Subtask utilizes ultrasensitive O_2 consumption (UOC) measurements at high temperatures to first confirm the expected correlation with high-temperature aging derived elongation data and then utilizes the sensitivity of UOC measurements to access temperatures down to actual service conditions in order to quantitatively test/confirm the Arrhenius extrapolation assumption. Results will allow much more confident predictions of cable material lifetimes. FY02 funding will be used to complete the elongation and UOC measurements on several important cable materials acquired in calendar years 2001-2002. Coupled with the results expected to be completed with FY01 funds, a large selection of important cable materials (e.g., several Hypalon and neoprene jackets, several CLPO and EPR cable insulations) will have been tested. Based on the Arrhenius studies of elongation coupled with the UOC results, conclusions about generic behaviors for important cable materials (e.g., Hypalon cable jackets) will be available, which will significantly reduce uncertainties related cable issues.

Subtask 2. Characterize materials that exhibit inverse temperature/Lazarus effects (ongoing in FY2002).
In a combined radiation/temperature environment, important cable materials (e.g., several CLPO and EPR insulations) age more rapidly at lower temperatures (e.g., contrary to the model associated with the Arrhenius equation). Some materials exhibiting inverse temperature behavior appear to include a secondary behavior that has been termed the "Lazarus" effect. When these materials are exposed to a high temperature after being aged, their mechanical properties improve dramatically. While exposure to high temperature would not be planned or credited in the cable qualification, the early stages of a loss-of-coolant accident (LOCA) could actually improve the material condition of these cables versus the common perception that a LOCA is the

sternest test of a material. When such effects exist, an alternative aging model must be developed. This Subtask utilizes various analytical approaches to understand the underlying mechanisms responsible for the unusual behaviors. To determine the breadth of materials having such effects, additional aging exposures are being made versus temperature at constant dose rate for a number of additional CLPO and EPR materials. The ultimate goal is to develop a viable predictive aging model for such materials.

Subtask 3. Evaluation/development of Wear-out methodology (ongoing in FY2002). The Wear-out approach is a novel method for predicting residual material lifetimes that was originally proposed and developed on weapon materials. In this method, pieces of material that have aged for long periods of time under ambient field conditions are subjected to an accelerated "Wear-out" temperature to drive the material to its "failure" condition. When time-temperature superposition is valid from ambient temperature through the Wear-out temperature, a linear relationship is predicted to occur between real time aging exposure and the time to failure at the Wear-out temperature. A plot of this relationship therefore allows an estimate of material lifetime under ambient conditions, offering an alternative means of periodically checking the predictions available from the accelerated extrapolations. The method is critically important for materials that show little evidence of damage with aging time until they catastrophically fail (referred to as "induction-time" behavior), since the Wear-out approach should transform such non-predictive behavior into linear predictive behavior. Evidence of "induction-time" behavior exists for many important nuclear power plant cable materials (e.g., many EPR and CLPO cable insulation). The expectation that non-predictive "induction-time" behavior would transform into a linear Wear-out plot has been confirmed from early screening results for an EPR cable insulation. Given this early success, the Wear-out approach will be applied/ developed/ optimized on various thermally aged cable materials with the focus on materials that have been oven-aged for extended periods of time (up to 8 years). The approach will also be applied to cables aged for up to 25 years in nuclear power plant environments (e.g., various cables from Duke Power as part of a proposed Duke pilot program to test both the Wear-out approach and various condition monitoring methods from Project 3-8). Eventual application of the Wear-out approach to condition monitoring (CM) samples is clearly desirable since CM samples could then be used to both determine cable condition and to predict remaining lifetime (both of which directly test accelerated extrapolations). For this reason, degradation during Wear-out exposures will be followed using candidate CM parameters (e.g., modulus, density, NMR- see Project 3-8).

Expected Duration: 3 years (FY 2000 – FY 2002)

FY02 Deliverables:

Several journal articles/reports will be written with at least one covering each of the three major Subtasks.

FY02 Estimated Cost: $330K

Total Estimated Cost: $1,235K

3-8 Develop Condition Monitoring (CM) Techniques and Database for Electrical Cable

Principle Objective: Develop nondestructive or essentially-nondestructive, science-based CM techniques for electrical cable insulation and jacket materials that are capable of characterizing the current condition of either a local section or an entire cable run using parameters (e.g., density, modulus, NMR relaxation time) correlated to traditional macroscopic degradation parameters (e.g., tensile elongation) or other well-defined criteria. The results of the effort will be included in the CM database

Need: The CM techniques developed will not only lead to more confident assessments of cable condition, but will also offer sample degradation monitors that will allow residual lifetimes to be estimated from the Wear-out approach (Project 3-7) on small material samples. The data from the effort will be placed in the Cable Condition Monitoring Database. The task results will allow more informed cable replacement decisions.

Expected Duration: 3 years (FY2000 – FY 2002)

Scope of Work:

FY2002 Scope of Work:

3-8.4 Develop Condition Monitoring Techniques for Electrical Cable

Investigate Advanced CM Techniques (Modulus Profiling, Density and NMR) (on-going in FY2002). The goals of this task include a careful and systematic investigation of the correlation of elongation results with data taken from several promising CM techniques including modulus profiling, density and NMR. Each of these CM approaches offers the potential for making measurements on very small samples (< 1 mg), much smaller than the size required for other candidate CM techniques. In addition, the small sample requirements means that the CM samples could also be used as a way of estimating residual cable lifetimes using the Wear-out method being developed in Project 3-7. Initial measurements for this ongoing project are concentrating on nuclear power plant cable jacket and insulation materials that have already undergone aging and mechanical property measurements (~3000 different combinations of materials and aging conditions are available). FY02 effort will focus on 1) most promising CM techniques (probably material dependent), 2) results on newly aged cable materials from on-going aging experiments in Project 3-7 and 3) optimization of CM approaches to small samples (<1 mg). For materials and environments where a particular CM approach appears to be promising, further studies will be conducted to determine how the correlation depends on environmental stress level (e.g., on the temperature or radiation dose rate). This is critically important since the accelerated stress levels where the correlation is derived will be much higher than the stress levels appropriate under the ambient aging conditions of interest to the eventual application of CM techniques. As part of this task, Sandia’s data (several thousand aging conditions) are being entered into a cable CM database.

Modulus Profiling- Results so far for thermal aging of an EPR insulation, a neoprene cable jacket and several Hypalon cable jackets indicate a remarkable temperature-independent correlation between modulus profile results and tensile elongation (the traditional standard method for monitoring cable material degradation). Even more remarkable is the observation that, for all of these materials, exceeding a modulus value of ~40 MPa corresponds to the absolute elongation dropping below ~30-50%. If similar results occur for materials from additional manufacturers and in additional environments, the 40 MPa level might represent a universal degradation benchmark, whether used as a CM benchmark or as a “failure-time” marker for Wear-out experiments (Project 3-7). Since the modulus profile apparatus is capable of measuring modulus with ~50-micrometer (2-mil) resolution, studies of very thin sacrificial slices from the outside surfaces of cables should be possible. Systematic, continuing studies in FY02 will focus on additional materials (from historic studies and from on-going aging studies from Project 3-7), additional environments and the optimization to small samples.

Density- Screening studies continue to show that drops in tensile elongation for most cable materials correlate with increases in density due to oxidation processes. The amount of increase is material and environment dependent. In thermal-only environments, density is relatively insensitive for CLPO insulation, but much more sensitive for EPR insulation and for neoprene and Hypalon jackets. The biggest percentage changes in density occur for EPR insulation near tensile failure. Since this rapid increase in density occurs at the point where this material fails quickly from a tensile elongation point-of-view (“induction-time” behavior), density measurements are ideal for predicting residual lifetimes from the Wear-out approach (Project 3-7) and will therefore be used (FY02) in this way for EPR insulation. Further development/ screening of density measurements will be done in FY02 concentrating on additional materials (from historic studies and from on-going aging studies from Project 3-7), additional environments and the optimization to small samples.

NMR- This method is based on increasing the sensitivity of NMR relaxation measurements by swelling the sample in a suitable solvent. Measurements are easily done on many commercial NMR machines, are extremely reproducible and typically require less than 15 to 20 minutes for sample preparation, data accumulation and data analysis. Screening studies of thermally aged materials indicate that the NMR approach is applicable to most important cable materials. This applicability includes CLPO insulation that are generally difficult to evaluate with most available CM techniques, because their material properties are dominated by the presence of a high modulus crystalline phase. Since NMR relaxation times are sensitive to the crosslink density in the amorphous phase of a material, the confounding effects of the crystallites on mechanical properties are eliminated. For example, studies of thermally aged Brandrex CLPO insulation show an excellent correlation between percent elongation and NMR relaxation times. Since results have been demonstrated on samples as small as 0.1 mg, the NMR approach is essentially non-destructive. In addition, the miniscule samples needed should prove useful for assaying the condition of samples that would otherwise be difficult to study such as composite thin insulation. Further development/screening of the NMR approach will be done in FY02 concentrating on additional materials (from historic studies and from on-going aging studies from Project 3-7) and additional environments. To optimize the measurements for each material type and environment, we will examine the effect of solvent type, measurement temperature and polymer concentration on the relationship between NMR relaxation times and mechanical properties.

3-8.5 Cable Condition Monitoring Database

The Sandia data generated in FY2002 and any additional data generated by industry in FY2002 will be gathered, formatted and included in the Cable CM Database. Manipulation and evaluation techniques will be finalized.

FY02 Deliverables:

- 3-8.4 Develop Condition Monitoring Techniques for Electrical Cable

Several journal articles/ reports will be written on the applicability to nuclear power plant cable materials of the CM techniques under study (e.g., modulus profiling, density, NMR relaxation).

- 3-8.5 Cable Condition Monitoring Database

An update to the Cable CM Database will be readied for use including FY2002 data.

FY02 Estimated Cost: $335K

Total Estimated Cost: $1,531K

3-13 Mechanical Behavior of Irradiated Structural Stainless Steels

Principal Objective: One of the objectives of the work is to determine the mechanical behavior of irradiated structural stainless steels under conditions of interest to LWRs and to develop constitutive models describing the behavior that can be used to develop tools to predict component life, assess the results of NDE examinations and guide the timing of corrective actions. A second objective is to determine the effect of irradiation history on the irradiation assisted stress corrosion behavior of multiple alloys of austenitic stainless steel and multiple heats of selected materials in PWR water. The third objective is to characterize mechanical and microstructural material properties of US PWR reactor internal materials irradiated in the Boris-6 fast reactor to provide data to correlate with modeling development.

The modeling effort will provide understanding and predictive tools that can then be used to project materials behavior and be used to manage plant aging. Operationally, the models will provide a means to predict component life, assess the results of NDE examinations, and guide the timing of corrective actions.

Need: Stainless steels, from which a number of reactor pressure-vessel internal components are made, experience a wide variety of material property degradation caused by exposure to the reactor-core operating environment. Among these types of degradation are an increased susceptibility to stress-corrosion cracking (SCC), a reduction in toughness and ductility, a reduced resistance to fatigue cracking, grain-boundary weakening, a reduced ability to relax residual stress concentrations, and a susceptibility to irradiation-induced dimensional changes caused by swelling and creep. These deleterious effects are all influenced by the mechanical behavior of the steels and the way the microstructure evolves during in-reactor thermal and irradiation exposure over a wide range of temperature and irradiation dose. The same irradiation-induced effects also affect the response to and effectiveness of repair processes such as welding. Therefore, understanding the changes that occur during irradiation of stainless steels is a critical aspect of understanding the degradation and repair of the reactor internals. Because all of the bulk mechanical properties are controlled by the microstructural changes that occur during reactor operation, the study of mechanical properties needs to be accompanied by a detailed microstructural characterization.

Most of the available data on the mechanical behavior of stainless steels under irradiation was developed in support of fast reactors, which operate at higher temperatures than LWRs. Considering the current desires for extended LWR lifetimes, ongoing programs are developing information on the effect of radiation on mechanical properties using several types of reactor (breeder, power, test) materials. The goal is to help develop a more complete understanding of the deformation behavior of these materials: their strength, post-yield-flow and creep properties as a function of strain rate, temperature, and irradiation history. The ongoing programs also include research on irradiation-assisted stress corrosion cracking (IASCC), intergranular stress corrosion cracking (IGSCC), and helium embrittlement, that will assist in the development of tools to predict component life, assess the results of NDE examinations and guide the timing of corrective actions. The information will also be relevant to the welding of irradiated materials.

The properties of structural austenitic stainless steels such as Type SA-304L (solution annealed), CW-316 (cold worked), SA-347, 308 (weld metal), and other potential replacement alloys for use as baffle/former bolts or other internals components need to be evaluated for their response to irradiation. The results of mechanical property tests will be used to evaluate the potential that irradiation embrittlement may cause premature failure of stainless steel internal components under design loading. Fractographic and microscopic examinations are necessary to develop correlation between the mechanical properties and fracture mode and the microstructural changes such as dislocation loops, bubbles, voids, and radiation induced segregation. These correlation need to be developed as a function of alloy, irradiation temperature, flux and dose.

PWR reactor internals materials irradiated to high dpa are needed for characterizing irradiation effects on material mechanical properties and understanding irradiation induced degradation and for benchmarking deformation model development.

Expected Duration: 6 years

Scope of Work:

Scope under EPRI funding

In conjunction with the three owners groups for PWR operating reactors in the US (WOG, B&WOG and CEOG) EPRI is supporting programs relevant to operating PWRs through the PWR Material Reliability Project (MRP). EPRI is also supporting research and development programs in progress at Electricite de France (EDF) that are directed at determining the effects of irradiation on the mechanical and stress corrosion behavior of the materials used in baffle/former bolts.

Structural austenitic stainless steels such as Type SA-304L (solution annealed), CW-316 (cold worked), SA-374, 308 (weld metal), and other potential replacement alloys for use as baffle/former bolts or other internals components are being irradiated to doses up to 80 dpa. Additional materials are added in the Bor-60 reactor in 2001 as shown in the attached table to expand the database. After irradiation, the materials will be subject to tensile tests. Fractographic and microscopic examinations will develop a correlation between the properties as a function of alloy; the irradiation temperature, flux and dose; and the microstructural changes (loops, bubbles, voids, etc.) and radiation induced segregation. Pressurized tube irradiation creep tests are also planned. Swelling mandrel tests in PWR environment and inert environment will be performed in Osiris at doses up to 12 dpa. Additional corrosion tests in PWR environment will be performed on selected materials. The effects of low helium level (about 30-75 ppm) and high helium level (about 300-750 ppm) on tensile and corrosion properties will be determined.

Three US PWRs removed baffle/former bolts during outages in 1998. Selected bolts were shipped to the Westinghouse hot cells for detailed examination and testing. The materials of the bolts are Type CW-316 and SA 347. Tensile, hardness, toughness, stress corrosion tests will be performed. Materials from other operating reactors will also be obtained and studied.
Actual core shroud material removed from a US BWR will be obtained for the fabrication of fracture toughness specimens. The results of the tests will be used to assess the implications for

BWR internals evaluations. Although at low fluences relative to PWRs, the information obtained will assist in the evaluation of the properties in PWRs.

The results of the mechanical property tests will be used to evaluate the potential that irradiation embrittlement may cause premature failure stainless steel internal components under design loadings. If the potential that irradiation embrittlement will reduce the fracture toughness below acceptable levels a plan will be developed to identify the critical locations for embrittlement and develop a plan for managing this aging effect.

High strength internals bolts (Alloys X-750 and A-286) are being evaluated for resistance to IASCC in an operating PWR. The bolts were preloaded to the recommended clamp loads on a test fixture that was paced in one of the surveillance capsule locations. At the completion of irradiation the bolts will be inspected for the presence of stress corrosion cracks.

Scope under DOE funding

Through a coordinated program of mechanical testing and microstructural examination of unirradiated and irradiated structural stainless steels and theoretical analysis of the results, the mechanical behavior of irradiated stainless steels will be determined and broadly applicable models for the deformation behavior of these materials will be developed. Argonne National Laboratory possesses an inventory of neutron-irradiated austenitic stainless steels and nickel-base alloys from EBR-II components, along with the corresponding unirradiated sibling material. These valuable materials will be used to study the effects of irradiation on the mechanical behavior of these alloys. To provide the broadest basis of experimental data for the modeling efforts, the intent is to coordinate with EPRI to supplement the EBR-II material with material irradiated in the BOR-60 reactor in Russia as part of the EPRI JOBB or CIR programs.

The project will be performing the mechanical testing of irradiated materials, primarily those machined from materials irradiated in EBR-II. Constant (slow) extension-rate testing (CERT), demonstrated on cold-worked materials in the previous year, will be used on irradiated 20% cold-worked 316 stainless steel and irradiated SA 304 stainless steel.

In addition, the mechanical testing will involve shear punch and automated ball indent testing on a variety of stainless steels which had been irradiated under varying dose rate conditions and to various total doses. Standard tensile testing of sheet or round specimens will be done as well to better benchmark the other test methods.

Transmission electron microscopy (TEM) samples will be prepared and analyzed with irradiation conditions corresponding to the tensile samples. As part of this materials characterization task, annealing studies will be done to prescribe a heat treatment that will help identify and quantify components of irradiation assisted stress corrosion sensitivity. The idea is to normalize irradiation-induced chemical segregation while leaving behind as many irradiation-hardening artifacts (dislocations, voids, etc.) as possible. Tensile specimens will be subjected to the heat treatment and subject to mechanical testing at slow strain rate. The results will be especially useful to associated modeling efforts.

As tensile test data and microstructural examination are completed, the model of deformation of irradiated materials will be refined.

The modeling in this project will first consist of empirical models that describe the effects of irradiation on mechanical properties and deformation/fracture behavior for austenitic stainless steels. This type of modeling will result in simple equations, maps, or graphs that explain the properties and the accuracy of the predictions. Additionally, a model framework will be developed that first predicts the microstructural development under irradiation and then predicts the mechanical property changes based on the microstructural changes. Although the model framework is not expected to be robust enough to predict all mechanical property changes under irradiation for any LWR relevant irradiation condition, the framework can be continually updated as more data becomes available through ERPI, DOE, or other open literature sources.

Funded by JOBB and CIR, material samples including tensile specimens, o-ring specimens, and 3 mm diameter disks are being irradiated in Bor-60. Part of the specimens will be taken out in 2002 which will have reached about 20 dpa irradiation and part of the specimens will be taken out in 2003 with about 40 dpa irradiation. These specimens are planned for testing starting in 2003 and through 2005.

Project Tasks:

<u>Tasks under EPRI funding – 3-13.1</u>

<u>Mechanical Behavior Studies:</u>

- Bor-60 Irradiation and Testing (EDF/JOBB Task B1)
  This task provides for the irradiation of austenitic stainless steels typical of those used in French PWRs or possible replacement alloys in the Russian Bor-60 reactor to exposures up to 80 dpa. This task also includes the mechanical testing and evaluation of the materials.
- Helium Effect Evaluation (EDF/JOBB Task B3)
  This task provides for irradiation of materials in the Russian SM reactor and for mechanical testing to assess the influence of helium resulting form transmutation reactions on the mechanical behavior of SA-304L and CW-316 stainless steel.
- EBR II Irradiation B Mechanical Testing and Microstructural Examination (EDF/JOBB Task B4)
  This task provides for the documentation of the previous EDF effort for the evaluation of the effects of irradiation in a fast breeder reactor. Microstructural evaluation of the irradiated materials.
- Specific Tests on EPRI Materials (EDF/JOBB Task B8)
  Austenitic stainless steel materials from US reactors will be irradiated as part of EDF/JOBB Task B1.
  This task will evaluate the effects of irradiation on the mechanical properties and the stress corrosion properties of the materials after irradiation up to 20 dpa.
- Characterization of Irradiated Materials (EDF/JOBB Task B5)
  This task will evaluate the microstructure and the radiation induced segregation of representative 304L and 316 stainless steels and other alloys from irradiations in Bor-60, EBR-II, SM, and Osiris.

- Irradiation Embrittlement of Reactor Vessel (RI-ITG Task 3.7)
  Internals in PWRs.
  This task will review existing information on irradiation embrittlement and prepare a white paper on its effects on PWR reactor vessel internals components. Based upon the white paper, PWR internals will be evaluated and ranked, by fluence and possibly other criteria, to determine the critical locations for irradiation embrittlement.
- Fracture Toughness of Irradiated Stainless Steels (BWRVIP Task 2.2)
  This task will obtain irradiated core shroud material removed from a US BWR and perform fracture toughness tests. The goal is to demonstrate that BWR core internals have adequate fracture toughness and that the fluence threshold can be extended

Corrosion studies:

- Hot Cell Material Testing of Baffle/Former Bolts (RI-ITG Task 3.2)
  Removed from Two Lead Plants
  This task will perform SSRT (IASCC) and IASCC crack growth tests on CW 316 and SA 347 baffle bolts removed from operating PWRs. Additional tensile and toughness tests are also to be conducted.
- Corrosion Tests on Chooz A baffle plate (EDF/JOBB Task B2a)
  This task will perform stress corrosion tests on irradiated (23 to 32 dpa) SA304 SS from a Chooz A baffle plate.
- Osiris Irradiation in PWR and Inert Environment (EDF/JOBB Task B6)
  Type SA-304L and CW-316 stainless steel will be irradiated and tested in core as expanding mandrel samples in inert and PWR water to study IASCC.
- SCC of High Strength Reactor Vessel Internals (RI-ITG Task 3.6)
  Bolting in PWRs
  This task will perform visual and UT inspections of Alloy X-750 and A-286 bolts assembled in a capsule and irradiated in an operating PWR. The bolts will be evaluated for the presence of stress corrosion cracks. Analyses of the results will help determine the effect on long term operation of these materials.
- Evaluation of the Effects of Irradiation on the IASCC (RI-ITG Task 3.8)
  and Mechanical Properties of Core Shroud Materials (SA 316, SA 304, CW 304, and 308 welds)
  This task will evaluate the effects of long term irradiation (up to 80 dpa) on the IASCC resistance of the austenitic stainless steel alloys used in US PWRs. It is proposed that the irradiation be performed in the Bor-60 and the results compared to the EDF/JOBB program results.
- Support for the International IASCC Program, Phase 2 (RI-ITG Task 3.13)
  This task provides support to an international program that will test and evaluate the effects of irradiation on the IASCC resistance of test materials and materials from actual in-service hardware with high exposure.

Tasks under DOE funding -- 3-13.2

- Prepare samples from hardware and surveillance samples retrieved from EBR-II. This task involves in-hot-cell milling, punching, grinding, and EDM preparation of irradiated material.

The samples prepared in this task will include tensile and TEM samples to be analyzed throughout the follow-on years of this program. (2001 Task)
- Perform static tensile tests on neutron-irradiated austenitic stainless steels from EBR-II. Perform mechanical testing on samples irradiated in BOR-60 as part of the JOBB and CIR program irradiations. Perform microstructural (TEM) examinations of irradiated steels. (2001-04 Task)
- Extend deformation model to describe irradiation effects. (2001-04 Task)
- Develop capability to perform microsample tensile tests on austenitic stainless steels using shear punch and automated ball indentation techniques. (2001-04 Task)
- Strain the tensile samples at a low strain rate (10^{-7} sec^{-1}) to failure. Perform fractography and microscopy studies. Perform microstructural (TEM) examinations of irradiated and deformed steels. (2002-04 Task)
- Perform annealing studies to eliminate grain boundary impurity segregation but retain the void microstructure. Perform microstructural (TEM) examinations of irradiated and annealed steels. Prepare tensile sample, anneal and strain to fracture to study the specific isolated microstructural variables that may affect IASCC. Perform small sample testing on annealed material. (2003-04 Task)
- Starting in FY2001, for each year of the project, student participation in these efforts by faculty or students from minority educational institutes, will be sought. (2001-05 Task)

Tasks under DOE funding -- 3-13.3

- Test material samples including tensile specimens, o-ring specimens, and 3 mm diameter disks that are being irradiated in Bor-60. (2003-05 Task)
- Interpret the test results, benchmark the model, and modify the model as needed. (2003-05 Task)

Funding:

FY02 Estimated Cost:	DOE: $465K	EPRI: $800K
Total Estimated Cost:	DOE: $5,085K	EPRI: $11,301 K

3-24 Fatigue

Principal Objective: The principal objective of this work is to provide cost effective methods of evaluating the cyclic life of nuclear components, including the effects of reactor coolant environment, and to provide utilities with effective tools to manage fatigue sensitive locations in Class 1 piping and components with explicit consideration of thermal fatigue.

Need: Cracking due to thermal fatigue resulting from cyclic thermal stratification has occurred in PWR RCS attached piping. Operating experience has shown that there are thermal fatigue transients which were not properly captured in the original fatigue design basis. The EPRI PWR MRP established a Fatigue Issue Task Group to address this issue and to provide utilities with a set of tools to manage thermal fatigue. One of the tools being developed by the MRP is a Thermal Fatigue Screening Tool. The ability to screen RCS components for susceptibility to thermal fatigue is needed in order to effectively manage this issue in a cost-effective manner.

Data has been generated that suggests a reduction in component fatigue life when reactor water effects are considered. However, laboratory data projections have not been supported by plant operating experience. Industry activities have recommended that existing laboratory data be reviewed regarding relevance to plant operating conditions and that additional laboratory data be generated under typical operating conditions and benchmarked against full-scale component testing. The resolution of this issue is necessary to provide appropriate guidance for Code committees action and may affect future design criteria regarding fatigue and present operating plants during license renewal. In addition, laboratory tests have suggested a reduction in fatigue life due to the affects of reactor water environment. A management strategy for considering reactor water effects on component fatigue life is needed to provide guidance for utilities seeking extended operation.

Expected Duration: 5 years (FY 2000- FY 2004)

Scope of Work:

Task 3-24 is an ongoing activity, organized into several subtasks, that was funded by industry and DOE NEPO in FY00 and FY01. Proposed tasks for FY2002, and the corresponding subtasks 3-24.2, 3-24.3, and 3-24.7, are described below. For continuity other subtasks, for which NEPO FY2002 funding is not being pursued, are shown.

<u>FY2002 Scope of Work:</u>

Subtask 3-24.1 Thermal Fatigue Management (EPRI funded in FY02)

Subtask 3-24.2 Thermal Fatigue Screening Tool

Task 1. Screening Model Development (Funded in FY00 and FY01)
Task 2. Screening Tool Development

The results from previous activities performed under this subtask will be extended to various piping configurations and geometries. A methodology will be developed (technical report and/or a software tool) to determine where potentially significant thermal fatigue damage may occur in PWR Reactor Coolant System attached piping.

Subtask 3-24.3 Fatigue Reactor Water Environmental Effects

Task 1. Review of Fatigue Environmental Data (Funded in FY00)

Task 2. Laboratory Tests

This task, initiated with NEPO FY01 funding, will generate experimental data under realistic geometric configurations to evaluate the reduction in fatigue life due to reactor water environment. Environmental conditions during testing will be specifically controlled to approximate those anticipated during service. Results will be compared to existing laboratory data and field experience to help determine impact of reactor water environment on fatigue life.

Subtask 3-24.5 Analytical Methods (EPRI funded in FY02)

Subtask 3-24.6 ASME Code Support (Funded in FY01. No funding necessary in FY02, possible funding in FY03)

Subtask 3-24.7 Fatigue Management

Task 1. Fatigue Lead Plant Inspection Program

The objective of this task is to define an in-service inspection program on fatigue sensitive Class 1 locations for which explicit consideration of fatigue reactor water environmental effects are expected to result in high fatigue usage factors. These locations would be the lead locations where cracking due to fatigue would be expected to occur. This task will identify those 'lead' locations and provide an estimate of fatigue usage with environmental effects, identify existing inspection information for those locations, identify utilities who would participate in this program, and develop a preliminary lead plant inspection program. Activities will be initiated in FY02 and completed in FY03.

Task 2. Fatigue Crack Frequency Evaluation (Funded by EPRI in FY01)

The Pacific Northwest National Laboratory study on "Evaluation of Environmental Effects on Fatigue Life of Piping" will be evaluated to identify significant conservatisms in this study. The PC-Praise code used in this study will be obtained and additional analyses with more realistic assumptions will be performed. The objective is to obtain additional data on best estimates of potential pipe crack frequency due to fatigue, including reactor water environmental effects. Activities will be initiated in FY01 and continue into FY02.

Task 3. Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (Funded in FY00 and FY01)

Task 4. Fatigue Technical Working Group (EPRI Funded)

Task 5. International Fatigue Test Program (Partially Funded in FY01)

This subtask supports DOE and EPRI participation in fatigue research programs being performed by EDF and various Japanese utilities, concerning thermal fatigue and reactor water environmental effects. The objectives of this program are to obtain (1) basic data on the high cycle fatigue behavior of 304L and 316L stainless steels (EDF), (2) in-plant thermal fatigue monitoring data at EDF PWRs, (3) thermal fatigue mixing tee tests and experiments (EDF)(4) results of thermal fatigue laboratory experiments performed by EDF and MHI in Japan, (5)results of fatigue reactor water environmental tests performed by IHI in Japan, and (6) results from the newly formed JSME Joint PWR and BWR Thermal Fatigue Program.

FY02 Deliverables:

- <u>Subtask 3-24.2</u>
  Thermal fatigue screening methodology and/or software tool (Subtask 3-24.2)

- Subtask 3-24.3
  Funding provided in FY01 is being utilized to develop a detailed test plan and to initiate testing activities. FY02 funding will be utilized to complete testing activities and develop a technical report that provides comprehensive results of all testing and reconciles the results with the report generated under FY01 funding. (Subtask 3-24.3)
- Subtask 3-24.7
  A progress report will be prepared that summarizes the lead components locations to be considered in an inspection program and provides an assessment of the effects of reactor water environment. The lead plant inspection program will be included in the final document to be prepared in FY03.
  A technical report will be prepared that summarizes all activities under the Fatigue Crack Frequency Evaluation task.
  A technical report will be prepared that evaluates the data obtained under the International Fatigue Test Program

FY02 Estimated Cost: $660 K

Total Estimated Cost: $2,320 K

3-27 Assessment of Aging Effects on Components and Structures from Nuclear Power Plants

Principle Objective: Obtain materials and components that have been in service in operating reactors to be used for comparison with laboratory aged materials to validate models for aging effects and nondestructive examination methods. Provide information on the significance of aging effects and the effectiveness of plant programs for managing aging effects..

Need: Data on the effects of component aging is required to support extended plant life and is typically estimated through laboratory testing, test reactor studies, or analytical model prediction. Obtaining 40-year equivalent aging data to support plant licensing is typically done in accelerated aging tests, using artificially high temperatures or radiation dose rates and extrapolating short duration test data. The data from these accelerated aging tests help to explain and quantify irradiation induced degradation mechanisms such as IASCC, void swelling, embrittlement, and stress relaxation. These accelerated aging results need to be "benchmarked" against components and materials that have been irradiated under prototypical PWR service conditions to support utilities' license renewal submittal for 60+ years of safe and reliable plant operation. To perform these "benchmark" tests, components from operational reactors need to be retrieved and tested. In addition to providing information on the effects of aging, components from operational reactors are also needed to validate new techniques for nondestructive material damage assessment.

The components required can be obtained from a number of sources. In some cases, the components of interest may have been replaced. In others they can be obtained from plants which have been retired from service. Other useful sources of materials may be in plant surveillance and monitoring programs for reactor vessel embrittlement or cable aging and components removed by utilities for failure analyses. In the past, opportunities to obtain components have been missed because of: the unavailability of resources, the lack of a systematic evaluation of the needs for component retrieval and assessment, and the relatively short "window of opportunity" to obtain components prior to disposal.

Scope of Work:

This project involves the following tasks: removal, storage, and testing of important components from operational plants, in-place examination of aging effects in components and structures for which there is limited operating plant inspection/assessment experience, and the development of a plan for obtaining important components from operational plants.

<u>FY2002 Scope of Work</u>:

Task 3-27.2 Acquire irradiated material from a decommissioned reactor (complete in FY2001)

The original scope of work for task 3-27.2 was to obtain core shroud material from the Maine Yankee reactor during decommissioning. Material was to be obtained using FY2000 funding of \$175K and a test plan written using FY2001 funding. Because scheduling difficulties precluded obtaining material from Maine Yankee, the focus shifted to obtaining material from the San Onofre Nuclear Generating Station Unit 1 (SONGS-1). Task descriptions including the test plan for San Onofre material have been moved to task 3-27.5

Task 3-27.3 Store irradiated material from a decommissioned reactor (to be redirected)

The original scope of work for task 3-27.3 was to store core shroud material from the Maine Yankee reactor acquired under Task 3-27.2. Material was to be stored in a National Laboratory using approved FY2000 funding of $50K. Because scheduling difficulties precluded obtaining material from Maine Yankee, the focus shifted to obtaining material from the San Onofre Nuclear Generating Station Unit 1 (SONGS-1). Since SONGS-1 materials are of Class C or greater, this task will have to be redirected. Task descriptions for San Onofre material have been moved to task 3-27.5.

Oak Ridge National Laboratory may accept SONGS-1 Class C material such as the Core Barrel, but not material greater than Class C such as the SONGS-1 Baffle Plate or the Former Plate if it is to be stored in a alpha-free hot cell.

ORNL may also accept SONGS-1 reactor pressure vessel (RPV) trepans for shipment in a DOT certified cask directly from San Onofre or Barnwell to ORNL.

Tasks 3-27.4. Identification and Retrieval of Naturally-Aged Materials and Components (complete in 2001)

The goal of this task is to develop a needs matrix which identifies data needs of the industry relative to component aging, to develop an integrated plan for obtaining the materials, and to implement the plan. The plan will identify and prioritize needs for such aged components removed from service and assess the suitability of candidate components and structures to address identified technical issues. The needs matrix will be completed using FY2000 funding, the integrated plan completed using FY2001 funding.

Task 3-27.5 Acquire and store irradiated material from the SONGS-1 reactor

This Task will acquire materials from SONGS-1. The materials will be stored until test samples can be made and tested. Four major projects will be accomplished under this task: material retrieval, material transportation, development of a test plan, and material testing.

- Material Retrieval. Retrieve reactor internals components from SONGS-1. The materials under evaluation to be retrieved and stored for testing from SONGS-1 are:

a) Baffle Plate -- 1.5 inch strip from top and bottom of section 3A. 2 strips ~1.5”x13”x0.5”
b) Former Plate -- 8”x5”x5” triangle
c) Core Barrel -- Section to be cut depends on weld. Removed slice needs to be sectioned to fit in cask. 8”x8”x1.12”
d) Baffle-Former Bolt Shanks – Several

Contractor(s) will be chosen to handle, ship, store, test, and dispose of the SONGS materials. The contractor must be able to meet the following requirements:

a) Accept class C or greater than Class C materials.
b) Handle large casks with underwater capability needed to retrieve samples.
c) Dispose of class C or greater than Class C materials.
d) Work with PCI, the contractor who is doing the cutting and dismantling of SONGS-1 components.
e) Be familiar with SONGS-1 design information.

Sample retrieval is a FY2000 task carried over from Task 3-27.2

- Transportation/Storage of Removed Samples

1. Arrange for and provide shipping cask for sample transportation meeting all applicable requirements.
2. Arrange for transportation of empty shipping cask to SONGS 1 and loaded cask to storage hot cells including all necessary shipping cask preparations/decontamination.
3. Provide hot cell storage for samples until testing program is initiated. Accept responsibility for disposal of all samples.

This is a FY2002 task since Task 3-27.3 is to be redirected.

- Test Plan. A test plan for the SONGS 1 samples will be developed with guidance from the EPRI MRP Internals Issue Task Group. The test plan is a FY2001 Task carried over from Task 3-27.2.

- Material Testing. Contractor(s) will be chosen to perform mechanical, microstructure, corrosion, fracture toughness and crack growth testing of retrieved samples including but not limited to the following:

a) Gamma Spectroscopy. 2002 Task
b) Manufacture Specimens and Conduct Tensile Tests. 2002 Task
c) Manufacture Specimens and Conduct IASCC Growth Testing. 2002 Task
d) Manufacture Specimens and Conduct Toughness Tests. 2002 Task
e) Manufacture and Test Slow Strain Rate Test (SSRT) Specimens. 2003 Task
f) Microstructural Examination. 2003 Task
g) Interpretation of Data. 2003 Task
h) Dispose SONGS-1 materials. 2003 and 2004 Tasks

FY02 Deliverable:
A report describing the results of the gamma spectroscopy, tensile tests, IASCC growth tests, and toughness tests.

FY02 Estimated Cost: $500K

Total Estimated Cost: $1,992K (combined tasks 3-27.3, 3 -27.4 and 3-27.5)

3-29 Motor Rewind Insulation System Development and Qualification for Harsh Environments

Principal Objective: To design and qualify motor insulation systems for use in harsh environment applications.

Need: Options for rewinding or replacing environmentally qualified (EQ) motors are extremely limited. Many original motor styles are no longer manufactured and equivalent replacements cannot be purchased. Installing new replacement motors of different design requires significant plant modifications. When available, rewinding options are limited to a couple of suppliers. Lead-times for manufacturing or rewinding certain large motors are a year or longer. Motor repair/replacement options are expected to decrease further as qualified insulating materials are no longer supplied, older motor designs become obsolete, and additional OEMs leave the nuclear industry. If not addressed, lack of EQ motor availability will have a significant impact on future nuclear plant operations, including extended plant shutdowns.

Expected Duration: 3 years (FY2001-FY2003)

Scope of work:

<u>FY2002 Scope of Work:</u>

<u>Random-wound Continuous Duty Motor Qualification for Inside Containment</u>
Random-wound motors fabricated in FY2001 Task 3-29.2 will be subjected to a nuclear harsh environment qualification test program in accordance with the test plan previously developed for the EPRI project. Work will be performed in accordance with 10 CFR 50 Appendix B QA requirements. Results of this task will be documented in a test report.

<u>Low-Voltage Form-Wound Qualification for Inside Containment</u>
Low-voltage form-wound test specimens previously manufactured by EPRI will be subjected to a nuclear harsh environment qualification test program in accordance with the test plan previously developed for the EPRI project. Work will be performed in accordance with 10 CFR 50 Appendix B QA requirements. Results of this task will be documented in a test report.

<u>Low-voltage Form-Wound Fabrication Procedures</u>
If necessary, fabrication procedures previously published by EPRI will be updated and revised to reflect the actual qualified motor fabrication methods and materials. This activity may not be required if the new qualification testing does not impact fabrication methods and materials qualified in previous EPRI work.

<u>Thermal class testing</u>
Insulation system thermal class testing will be performed on the random-wound and form-wound motor insulation systems. Thermal class testing results will allow utilities to optimize motor qualified lives.

Work Scope for Subsequent Periods:

Random-wound Continuous Duty Motor Fabrication Procedures
Preliminary fabrication procedures used to build the test specimens will be updated and revised to reflect the actual qualified motor fabrication methods and materials.

Final Project Report
The project team will produce a final project report documenting the EPRI and NEPO project results since project inception. The scope of the report will summarize material screening tests and selection, fabrication issues, qualification testing, thermal class testing, and material baseline information.

FY02 Deliverables:

- Random-wound continuous duty motor insulation system qualification test report
- Low-voltage form-wound motor insulation system qualification test report
- Low-voltage form-wound motor insulation system fabrication procedure (if necessary)
- Insulation system thermal class testing report

FY02 Estimated Cost: $390K

Total Estimated: $933K

3-30 Irradiation Induced Swelling and Irradiation Enhanced Stress Relaxation of PWR Reactor Core Internal Components

Principal Objective: Characterize irradiation induced void swelling and stress relaxation related degradation that could occur in operating reactors through in-situ measurement, and calibrate and extend the breeder reactor based swelling model for PWR applications.

Need: Utilities that operate Pressurized Water Reactors (PWRs) are concerned with the potential of core shroud components degradation by void induced swelling due to neutron irradiation. Void swelling of the core shroud panels/baffle plates could increase loads at the connection points (e.g., bolted or welded joints at the former plates and other structural members) and swelling above 5% may cause further embrittlement, potentially accelerate cracking (e.g., Irradiation Assisted Stress Corrosion Cracking) in service. Conversely, irradiation enhanced stress relaxation might act to mitigate the effect of the dimensional changes, relative to IASCC.

In addition, irradiation enhanced stress relaxation (SR) has been identified as a potential aging mechanism that may influence the service lives of reactor internals bolting. Data have shown that the thermal effects on annealed Type 304 stainless steel at temperatures below 900^{o}F produce a stress relaxation maximum of about 18%. The combined effects of temperature and irradiation have been reported to produce further relaxation. These data suggest that the combined effects of temperature and irradiation can result in significant stress relaxation of preloaded bolted connections. Therefore, irradiation enhanced stress relaxation (SR) is a potential concern for all bolting used in the RV internals.

Data from the components of PWR core internals with operating service experience to quantify irradiation induced swelling of PWR internals is limited, particularly measurements in actual operating plants. This type of information is needed. Existing equations based on breeder reactor data and experience need to be calibrated and extended to PWRs based on data obtained under PWR operating conditions. Information is being generated in several types of reactors (breeder, test, and operating) that will also be useful in validating the equation parameters.

Expected Duration: 4 years (FY 2000 – FY 2003)

Scope of Work:

<u>Scope under EPRI funding</u>

EPRI has an ongoing three-year effort to study void swelling, and stress relaxation of internals fasteners starting in 1999 in the PWR Material Reliability Project (MRP). The objective of the MRP effort is to evaluate the extent of void swelling and irradiation creep related degradation that could occur in PWR core internals materials as a result of exposure to neutron irradiation. This MRP effort will integrate the existing EDF/JOBB irradiation program with other programs to obtain data on high fluence exposure of plates, welds and bolting material. The available data on void swelling and stress relaxation due to irradiation will be reviewed. A report will be prepared that will include 1) guidelines for assessment of the potential for void swelling and irradiation enhanced stress relaxation for PWR applications, 2) engineering assessment of where void swelling and stress relaxation may be manifested, 3) an assessment by each of the team members representing the three primary US NSSS vendors (W, FTI, ABB/CE) to identify where and how the effects of void swelling and irradiation enhanced stress relaxation will be manifested, and 4) a program plan for obtaining new, relevant data, including possible in-situ measurement of in-service core internals. A listing of potential PWR sites for the measurements of reactor internals will be developed.

Scope under DOE funding

Supplementing the MRP studies, three activities are proposed in this project: 1) The development of NDE methods to measure swelling in the field. 2) The development and documentation of an equation, based on present day knowledge, to predict when, were and the magnitude of void swelling for PWR internals.. 3) The design of a NDE tool for detecting void swelling compatible with present day tooling used to detect baffle bolt cracking.

For the first activity, NDE techniques will be evaluated to determine the two best techniques for measuring percent swelling.

The developed techniques will be applied for measurements on selected thin and thick specimens with known swelling to assure reliable results. The developed technique and equipment with recommended procedures will be made available to the industry for examinations of reactors that have conditions that might produce significant swelling.

Equations for calculating and extrapolating swelling to the commercial PWR environment have been based primarily on the breeder reactor data. However, the breeder reactor temperature and flux spectrum, which are two key parameters in inducing swelling, are not the same as those in a PWR. PWR equations will be developed in parallel with the inspection techniques to provide guidance to determine when to inspect and at what locations. These equations will be updated as PWR data becomes available.

Project Tasks:

Tasks under EPRI funding – 3-30.1

- Prepare a technical basis document report outline on the evaluation of the extent of void swelling and stress relaxation related degradation that could occur in PWR core internals materials as a result of exposure to fast neutron irradiation. (2000 Task)
- Using the outline developed, prepare a detailed technical basis document report. The 2000 task is a set of guidelines for assessment of the potential for void swelling and irradiation enhanced stress relaxation for PWR applications, based on available data. (2000 Task)
- Prepare specific recommendations for implementation of guidelines developed in 2000 for assessing irradiation induced swelling and stress relaxation. These recommendations will be reviewed with utilities seeking their approval and define scope of work needed. (2001 Task)
- The available swelling and stress relaxation data will be collected and evaluated including those generated under the MRP sponsored PWR baffle bolt testing and high fluence thimble tube testing from a Swedish PWR plant. (2001 Task)

- The data evaluation results will be used to ascertain the significance of void swelling and irradiation enhanced stress relaxation and to provide recommendations, if needed, for managing the effects as an age related degradation mechanism. (2001 Task)
- Develop recommendations for obtaining additional, specific data on irradiation induced swelling and/or stress relaxation, including the need for in-situ measurements or examinations on in-service components. This task also includes the identification of potential operating reactors at which dimensional inspection of core internals may be performed. (2002 Task)
- Irradiate specimens in Bor-60 reactor for TEM testing to get swelling data up to 40 dpa or higher. (2001-2003 Task)
- Obtain swelling data from a thimble tube highly irradiated in an operating PWR. (2002-2002 Task)

Tasks under DOE funding – 3-30.2

- Test all candidate NDT techniques on non-radioactive surrogate materials and select most promising techniques for in-situ NDT void swelling measurements. (2000-2001 Task)
- Develop predictive equations for void swelling and irradiation creep to identify areas for in-situ testing. (2001 Task)
- Construct prototype devices for in-situ applications. (2001 Task)
- Conduct hot cell tests and experimental evaluation of selected sensors on both thin and thick radioactive materials. (2001 Task)
- Develop 3-D maps of void swelling for reactors chosen by the Advisors Group. (2001 Task)
- Continue hot cell tests and experimental evaluation of selected sensors on both thin and thick radioactive materials (using materials from SONGS-1 will be explored). (2002 Task)
- Refine the design and the mounting of the selected devices. (2002 Task)
- Select reactors that are good candidates for in-situ measurement based on 3-D mapping information. (2002 Task)
- Conduct in-situ measurement in selected reactor(s). (2002-2003 Task)
- Evaluate the in-plant data to extract fundamental mechanism information for irradiation induced swelling. (2003 Task)
- Evaluate the boundary constraining effect on in-plant component swelling. (2003 Task)
- Extend and modify PWR swelling equations based on in-plant data and all relevant data available in industry for PWR applications. (2003 Task)

FY02 Estimated Cost: DOE: $400K EPRI: $240K

Total Estimated Cost: DOE: $1,850K EPRI: $1,468K

3-207 Mitigation of Initiation and Growth of PWSCC in Alloy 600 and 82/182 Weld Metals

Principle Objective: Develop mitigation methods to address the initiation and growth of PWSCC in Alloy 600 and weld metals 182 and 82 in PWR service.

Need: Laboratory testing results and field experience have demonstrated for many years (since the early 70's) that Alloy 600, in the form of wrought products such as steam generator tubing, plate or bar, and its weld metals 182 and 82 can experience primary water stress corrosion cracking (PWSCC). Recently, in service PWSCC of control rod drive mechanisms (CRDMs) at the Oconee units and butt welds of 82 and 182 in a reactor vessel (RV) hot leg nozzle at the V. C. Summer station, has heightened the interest of utilities to this issue. The interest is from both degradation management and safety assessment perspectives. From a degradation management perspective a utility would like to have an approach to know when and how to inspect components in a systematic and managed way in addition to having tools available to either repair and/or mitigate degradation. From a safety perspective, it is important to have an understanding of the rate of degradation, i.e. crack growth rate, to assure that the extent of degradation will never exceed conditions challenging structural integrity.

Expected Duration: 3 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

Task 1 – Test Matrix Development, materials acquisition and specimen preparation

- Test matrix development and identification mitigation strategies.
- Chemical composition analysis, mechanical properties, and microstructural characterization of materials.
- Selection of Alloys 600 and 690 material compositions and manufacturing of welds in Alloys 82, 182, 52, and 152.
- Selection of service exposed material [i.e. V. C. Summer(VCS) and Oconee Samples].
- Machining and pre-cracking of specimens (~100)

Task 2 – Accelerated and Prototypic Environment Crack Initiation Testing

Task 2A – Relative Ranking Tests– Doped Steam

- High stressed samples-triplicate replication
- All (including Alloys 690, 52 and 152 welds) but service materials

Task 2B – Relative Ranking Tests– Primary Water @ 330^{o}C

- High stressed samples-triplicate replication
- All materials, including service materials.

Task 2C – Mitigation Methods Testing– Doped Steam

- Only most susceptible material

- High stressed samples with mitigation applied
- Sets of samples at low, medium, and high stresses

Task 2D– Mitigation Methods Testing– Primary Water @ 330^{o}C
- Only most susceptible material
- High stressed samples with mitigation applied
- Sets of samples at low, medium, and high stresses

Task 3 – Crack Growth Testing in Primary Water @ 330^{o}C
Task 3A – Influence of Stress Intensity on Alloy 182 and 82 Welds, and VCS and Oconee materials
Task 3B – Influence of Zn, Li-B (pH), H_2 on Alloy 182 Welds at a single high stress intensity.
Task 3C – Long term tests of Alloy 690, 52, 152, and 82 Welds at a single high stress intensity.

Task 4 – Meetings, management, and reporting.

<u>Work Scope for 2003 - 2004</u>:

Task 2 – Accelerated and Prototypic Environment Crack Initiation Testing
Task 2A – Relative Ranking Tests– Doped Steam
Continuation of resistant materials.
Task 2B – Relative Ranking Tests– Primary Water @ 330^{o}C
Task 2D– Mitigation Methods Testing– Primary Water @ 330^{o}C

Task 3 – Crack Growth Testing in Primary Water @ 330^{o}C
Task 3A – Influence of Stress Intensity on Alloy 182 and 82 Welds, and VCS and Oconee materials
Task 3B – Influence of Zn, Li-B (pH), H_2 on Alloy 182 Welds at a single high stress intensity.
Task 3C – Continuation of long term tests of Alloy 690, 52, 152, and 82 welds at a single high stress intensity.

Task 4 – Meetings, management, and final reporting.

FY02 Deliverable:

Interim report of crack initiation test results in doped steam and primary water environments, and results of completed crack growth tests.

FY02 Estimated Cost: $125K

Total Estimated Cost: $575K

3-209 Validation of BWR Fluence Models and Weldability of Internals

Principal Objective: Obtain material samples from BWR internal components to be used to benchmark BWR fluence calculation methodologies and to provide additional data for determining weldability of BWR internal components

Need: Additional data are needed to benchmark fluence calculation methodologies because fluence models are not benchmarked beyond the active fuel height of the reactor core. Such data would be used to resolve calculational uncertainties associated with predicting fluences at these locations. The information is also needed to determine the viability of welding irradiated material at high-fluence locations. Small samples (a few grams) can be removed from selected components such as the upper and lower regions of the core shroud, top guide, core plate, and core spray system. The samples can be analyzed to determine fluence and helium and boron content. This information is crucial to assess both the capability of models to accurately predict fluence in these locations as well as determine if these materials are capable of being weld-repaired without any deleterious long-term effects. More accurate fluence models would significantly improve component condition assessment for current and extended operating terms.

Expected Duration: 1 year

Scope of Work:

<u>EPRI BWR Vessels and Internals Project (BWRVIP) FY2002 Related Scope of Work:</u>

This task will develop a methodology to calculate the fast and thermal neutron fluence in a BWR. The unique feature of this methodology is that it will be a full 3D deterministic approach capable of determining fluence at the surveillance capsule, the vessel wall and core shroud within the active fuel height as well as locations that are well outside the active fuel height and include such components as the core spray piping, the top guide and core plate. The methodology is to be in accordance with Regulatory Guide 1.190. This task is funded by EPRI and is described because of its relationship to the proposed NEPO task

<u>NEPO FY2002 Scope of Work</u>:

The scope of work of this activity is:

1. Modify sampling tooling previously developed by the BWR Vessel and Internals Project (BWRVIP) to accommodate obtaining samples from BWR core shroud and core spray locations
2. Obtain samples at one or more locations to

a. determine helium and boron content at the weld locations. Incorporate information into the BWRVIP reports on weldability developed by a complementary BWRVIP activity
b. determine the fluence of the specimens and incorporate results into complementary BWRVIP activity to develop a BWR fluence calculation methodology

Work Scope for Subsequent Periods:

None

FY02 Deliverable:
Report documenting helium and boron content and fluence of the samples.

FY02 Estimated Cost: DOE: $500 K EPRI: $500 K

Total Estimated Cost: DOE: $500 K EPRI: $500 K

3-210 Low Temperature Hydrogen Cracking of Ni-Base Alloys and Weld Metals

Principle Objective: Determine the significance of low temperature crack growth in nickel base Alloy 600 and weld metals 182 and 82 in PWR service.

Need: Primary water stress corrosion cracking (PWSCC) of Alloy 600 and its weld metals 182 and 82 have recently been experience in CRDMs at Oconee units and in reactor vessel (RV) hot leg nozzles at V. C. Summer station. Bettis has published laboratory results demonstrating that weld metal Alloy 82, and even replacement weld metal 52 and Alloy 690 are susceptible to reduction in toughness as a results of hydrogen induced cracking. Thus far, destructive examinations performed on Oconee CRDM and V. C. Summer weld metal samples have not indicated whether low temperature crack growth has been involved in the failures. However, the knowledge of what to look for and inspection methods needs to developed. The industry needs to first understand whether the laboratory observed cracking phenomenon can occur in PWR service. If it can occur in service, how much of the apparent crack growth between cycles comes at low temperature, and what change(s) in shut down practices could be implemented to minimize its contribution.

Expected Duration: 2 years

Scope of Work:

FY2002 Scope of Work:

Task 1 – Reproduction of the phenomenon

Measurement of J_{IC} under the same conditions as Bettis (water at 54°C and 150 cc hydrogen/kg compared with air).

Task 2 – Materials acquisition and specimen preparation

- Selection of material compositions and manufacturing of welds in Alloys 82, 182, 52, and 152.
- Machining and pre-cracking of specimens (~50)
- Chemical composition analysis of the products

Task 3 – Experimental Development

- Supplying and validation of load and opening sensors to operate in hot water up to 330°C.
- Modifications to autoclave system
- Validation of hydrogen measurements
- Validation if J_{IC} measurements procedure

Task 4 – Effect of Temperature

Measurement of J_{IC} as a function of hydrogen content for Alloy 182 to determine the embrittlement transition curve.

Task 5 – Effect of hydrogen content

Measurement of J_{IC} as a function of hydrogen content for 182.

Work Scope for Subsequent Periods:

Task 6 – Testing of other materials such as 82, 52, 152, and Alloy 690.

Task 7 – Testing in simulated HWC of BWR

Task 8- Fracture surface examination by SEM

Task 9 – Final report

FY02 Deliverables:

Interim report describing the replication of Bettis results, specimen preparation, equipment development, and influence of temperature and dissolved hydrogen.

FY02 Estimated Cost: $150K

Total Estimated Cost: $225K

3-211 Thermal Aging Embrittlement of PWR Metals

Principal Objective: Provide a source book of existing data and new test data on thermally aged materials removed from plants to support effective aging management throughout the remaining licensed term or an extended operating term.

Need: The EPRI Materials Reliability Project is summarizing available thermal aging data of metal alloys commonly used for key components in elevated temperature applications in PWRs in 2001. The results will evaluate aging embrittlement and determine its significance for both life attainment and license renewal. Existing data indicates that embrittlement of nuclear system materials due to long-term elevated-temperature exposure (away from the high fluence areas) is a potential aging mechanism that may influence the service lives of PWR components. In high fluence areas, the potential synergistic effect of temperature and fluence also requires assessment.

Expected Duration: 2 years

Scope of Work:

FY2002 Scope of Work:

- Complete a research summary on thermal aging data for materials selected from the following:
  - Austenitic stainless steels (Types 304, 316, 347, and XM-19)
  - Martensitic stainless steels (Types 403, 410, and 440)
  - Nickel-base alloys (Alloys 600 and 690)
  - Cast stainless steels (CF3, CF3M, CF8, and CF8M)
  - High strength bolting alloys (Alloys 17-4 PH, X-750, A-286, 718, SA-320 Gr L43 and SA-540 Gr B23/B24 Cl 3)
  - Low alloy steels (A 302 Gr B, A 508 Cl 2, A 533 Gr B, and A 516 Gr 70)
  - Weld alloys and weld joints for a variety of materials
- Review license renewal documents and interview NSSS vendors to ascertain data needs and availability of highly susceptible materials from operating or decommissioned plants.
- Prepare detailed test plans for materials recommended by EPRI, MRP, vendors, and DOE
- Obtain material specimens from operating and decommissioned plants
- Perform tests on removed specimens
- Incorporate existing data and the new test data into a Metals Thermal Embrittlement Source Book

Work Scope for Subsequent Periods:

If needed, obtain additional samples of materials from plants, perform testing, and include new results in a revision to the Metals Thermal Embritttlement Source Book

FY02 Deliverable:

A Thermal Aging Embrittlement Source Book documenting existing information regarding thermal aging of the key materials used in elevated temperature applications in PWRs. The Source Book will identify material and temperature combinations where thermal aging needs to be considered, as well as where it need not be considered. The Source Book will be organized in a manner easy to use by utility engineers, including models or methods for extrapolation of results to the times and temperatures of interest.

FY01 EPRI/MRP Cost: $146 K

FY02 Estimated Cost: DOE: $250K EPRI: $250K

Total Estimated Cost: DOE: $500K plus costs for additional materials/tests to be determined based on the results of the FY02 effort.

3-212 Aging Data for Long-term Reliability of Systems, Structures, and Components (SSCs)

Principal Objective: Provide generic "foundation" information and data, which plant engineers can use to generate long-range aging and obsolescence management plans ("LCM plans") for preventive maintenance, replacement, and/or redesign of plant-specific SSCs important to safety, reliability, and economics. Reliability improvements will benefit plants with both 40- and 60-year operating terms. The prototype sourcebooks being produced in 2001 under EPRI funding provide a common format and content for about 40 sourcebooks planned to be produced over the next few years.

Need: LCM planning begins by researching generic industry operating experience information such as aging mechanisms, obsolescence, and performance. Once the generic information has been compiled, plant-specific data and reliability/economic evaluations are performed by each plant. LCM planning for all important SSCs in the US can cost hundreds of millions of dollars from reduced cost of planning and avoidance of lost generation. Almost half this amount can be saved by having expert researchers provide generic information in the form of sourcebooks. The result will be more consistent and complete LCM plans than if each plant started from scratch.

Expected Duration: 3 years

Scope of Work:

<u>FY2002 Scope</u>:

Produce sourcebooks for three important SSCs selected by DOE, EPRI, and utility advisors (e.g. main steam, main feedwater, circulating water, and service water systems). The foundation data to be researched and assessed include:

- performance experience from nuclear and non-nuclear plants
- regulatory issues and requirements
- known issues such as material vulnerabilities and obsolescence
- typical current maintenance activities
- state-of-art preventive and predictive maintenance technologies
- Industry failure rate data as a function maintenance approach
- Predictions of long-term failure rates
- Aging assessments (typical environments, aging mechanisms/effects/failure modes, residual life, etc.)
- Suggested maintenance and aging management alternatives based on industry/expert consensus

Research sources include NPAR, NUREGs (e.g. GALL, national lab aging management reports), NRC Generic Communications, EPRI (License Renewal, PM Basis, NMAC, etc.).
The research information will have the industry-consistent format and contents established in the EPRI LCM Sourcebook Overview Report and two prototypes sourcebooks (instrument air and underground piping).

Work Scope for Subsequent Periods:

Produce 3 or 4 aging sourcebooks per year in 2003 and 2004. SSCs will be those prioritized by DOE, EPRI, and industry advisors. The sourcebooks provided by this project will comprise about a quarter of the total needed by the industry.

FY02 Deliverables:

Three aging sourcebooks.

FY02 Estimated Cost: $175K

Total Estimated Cost: $525K

3-218 Advanced Millimeter-Wave Sensor for Non-Contact Non-Destructive Evaluation of Cast Stainless Steel Pipes

Principal Objective: Develop and demonstrate an advanced sensor for evaluation of cast stainless steel components in service, using millimeter-wave technology.

Need: Challenge in evaluation of cast stainless steel is the grain size. Acoustic or ultrasonic or optical techniques could not be applied due to the large grain size and attenuation in service environments. In service evaluation of cast stainless steel components (especially pipes) will result in significant cost savings and support personnel safety. Robust nondestructive evaluation technique is needed. Millimeter waves are suitable for this application, as they are not sensitive to grain size less than milli/sub-millimeter resolution and other plant environment (dust, smoke, and moisture). Additionally, optical quality of surface is not required.

Millimeter-wave instrumentation is a proven technology for temperature measurements in harsh, high temperature environments, but not yet applied to HLW or LAW melters. It is a routine tool for experimental fusion reactor plasma core temperature measurements, and has been demonstrated for use in arc plasma furnaces for pyrometry. Millimeter-waves are also less effected by non-optical quality surfaces and thin surface coatings that would further attenuate infrared radiation. However, millimeter-waves are still short enough so that viewing beams can be collimated and/or focused for good spatial resolution. The following are the advantages of millimeter-wave sensor:

- Use millimeter-wave thermal and coherent radiation - 10 - 0.3 mm (30 - 1000 GHz)
- Robust in harsh environment reliable in the presence of dust, smoke, thin films on components, and non-optical surfaces makes use of refractory materials for waveguide and optics components
- New parameter measurement potential (discontinuity in cast stainless steel)
- Remote electronics easy interface with control outside radiation shielding

As the millimeter-waves do not penetrate the metallic medium greatly, the waves will be either propagated through the surface or through engineered waveguides made of cast stainless steel.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

A suitable waveguide will be selected for millimeter-wave application. Surface and other modes of propagation of millimeter-waves through model cast stainless steel samples will be examined. All required hardware would be designed and fabricated. Test parameters for a wide variety of cast stainless steel shapes will be determined. Model cast stainless steel materials will be tested for defects (cracks, pores, bubbles, swelling). Existing expertise in this field will be leveraged to develop this technology in a short turn around time.

Work Scope for Subsequent Periods

In the second year, the millimeter wave sensor technology will be demonstrated in a suitable facility.

FY02 Deliverables:

- Select and characterize suitable waveguides and mode of operation
- Design and fabricate millimeter-wave sensor system
- Complete testing of model cast stainless steel materials under a wide range of test conditions.

FY02 Estimated Cost: **$** 250K

Total Estimated Cost: **$** 500K

3-221 Synergistic Effects of Irradiation and Thermal Aging in Cast Stainless Steels

Principle Objective: The objective of this work is to determine the impact of the combined effects of irradiation and thermal aging on the fracture toughness of cast stainless steels. The NRC has suggested that there may be a synergistic effect of these two factors. There are a limited number of materials available that have service histories appropriate for evaluating this effect. The program will optimize the use of available materials and generate new materials for testing.

Need: Thermal embrittlement of these cast stainless steels in the reactor internals must be considered as a potential degradation mode in the analysis for license renewal. The most critical applications of these materials in PWRs are in the lower core support structure. The bulk of this structure is well removed from the core and should experience maximum temperatures of 290^oC and peak EOL fluences of approximately10^{19} n/cm^2 (E>0.1MeV). However portions of this structure near the lower core plate will experience much higher EOL fluences. There are also a limited number of cast stainless steel components in the upper internals which tend to experience higher operating temperatures, but lower neutron fluences. Current acceptance of these components for extended life applications are based on two observations:

1. Cast stainless steels with low ferrite contents have limited susceptibility to long term thermal aging.
2. The rate of embrittlement is extremely low at 290^oC.

The proposed program will examine the validity of this assumption.

There is a need to augment the limited number of available materials that have experienced a combination of thermal and radiation exposures relevant to this application. In the proposed program, representative cast stainless steel specimens will be irradiated in a specially constructed research capsule. It is anticipated that these materials will include CF8 castings with a range of ferrite contents and at least one pre-aged material. These materials would be irradiated over an 18 month period in the University of Michigan Research Reactor to a total fluence of 10^{19} n/cm^2 (E>0.1MeV). The feasibility of obtaining higher fluence either by initiating long term irradiations in commercial reactors or accelerated irradiations in test reactors will also be evaluated. Test data from will be combined with data generated by testing materials removed from operating reactors. The interpretation of the mechanical test results will require a model to relate the observations to the underlying metallurgical behavior. The model will build on existing models of thermal aging susceptibility and irradiation embrittlement.

Expected Duration: 3 Years

Scope of Work:

FY2002 Scope of Work:

Task 1: Cast Stainless Steel Irradiation. Charpy and fracture toughness specimens of up to six cast stainless steels will be irradiated in the University of Michigan reactor to a total accumulated fluence of $1x10^{19}$ n/cm^2 (E> 0.1 MeV) over an 18 month period.

Task 2: Feasibility Study for High Fluence Irradiation Capsule. A survey of existing reactor designs will be conducted to establish a target fluence for the irradiation program. Preliminary investigations indicate that the target fluence may be as high as 10^{22}n/cm^2 (E> 0.1 MeV). An appropriate irradiation facility will be recommended and an irradiation program designed.

Task 3: Testing of Irradiated Castings. Available cast materials from a decommissioned reactor or from components removed from an operating reactor will be tested. Impact and fracture toughness testing of the cast materials would be conducted under this task.

Task 4: Evaluation of Embrittlement Behavior. The potential for long term embrittlement of cast stainless steel internals will be conducted. A survey of materials and operating conditions will be conducted. A model of cast stainless steel behavior during long term irradiation will be developed

Work Scope for Subsequent Periods:

Task 5: Testing of Specimens From Irradiation Capsule. Assuming an eighteen month irradiation cycle, specimen testing could begin no earlier than the third year of the program.

Task 6: Implementation of High Fluence Irradiation Program. The scope of this task would be defined under Task 2.

Task 7: Evaluation of Supplemental Materials. Several potential sources of additional irradiated cast stainless steel have been suggested. If preliminary evaluations indicate that testing of these materials could help resolve outstanding issues, the material would be tested in the second year of the program.

FY02 Deliverables:

- Initiation of Cast Stainless Steel Irradiation
- Report on Embrittlement Evaluation and Feasibility Study
- Test Results from Irradiated Castings

FY02 Estimated Cost: $565K

Total Estimated Cost: $965K + Amounts TBD in Tasks 2 & 7

3-222 Develop a Predictive Model for Pitting Corrosion of Heat Exchanger (HX) Tubes

Principle Objective: Establish data and a methodology that can be used to predict the (non-MIC) pitting corrosion of common heat exchanger tubing materials as a function of appropriate water chemistry parameters and the heat exchanger operating conditions. For new heat exchangers, these results can be used to select the appropriate materials for a given application. For existing heat exchangers, these results can be used to prioritize inspections and predict the remaining life of the tubing, so that repairs and replacements can be proactively planned and scheduled.

Need: Many of the heat exchangers used in nuclear and fossil power plants (as well as in chemical, pulp and paper, and other types of process plants) use river, lake, ocean, cooling pond, or well water on the tube side. These heat exchangers are found in the safety and non-safety service water, circulating water, and fire protection systems. The raw water is often untreated, or minimally treated, and may contain significant amounts of chlorides and sulfides which are well known to cause pitting corrosion in certain materials, principally the copper bearing alloys. However, such copper bearing alloys are inexpensive, easy to fabricate, and have desirable heat transfer characteristics. Conversely, materials that are highly resistant to pitting corrosion, such as titanium and the super stainless alloys, are expensive, more difficult to fabricate, and with less desirable heat transfer characteristics.

For new or replacement heat exchangers, the industry need is technology to select the right material for the operating environment. For existing heat exchangers, the industry need is technology to understand how well the material will survive in its environment so that inspections, repairs, and replacements can be prioritized and proactively scheduled.

Expected Duration: 2 Years

Scope of Work:

<u>FY2002 Scope of Work</u>:

Research has shown that pitting corrosion will not occur when the corrosion potential is below the reactivation potential. Data shows that the reactivation potential is a function of water chemistry (chloride and sulfide concentrations) and temperature. We believe that an empirical model can be built using these parameters to predict the reactivation potential. This model can be developed from matrix of experimental data based on measuring the reactivation potential during cyclic polarization runs performed at defined temperatures, and at defined chloride and sulfide concentrations.

The model can then be used to develop pitting resistance curves that can identify:

- The material will definitely not pit
- The material will definitely pit, and
- The material may pit.

Materials that fall into the may pit category will be ranked as to susceptibility to pitting and time to pit initiation.

The work will be focused on pitting associated with non-biological impurities in the water, and will include both data generation and model development. Pitting associated with the life cycle of microorganisms (.e. g., microbiologically influenced corrosion - MIC) is outside the scope of this proposal.

Work Scope for Subsequent Periods:

The second phase (FY2003) will be to test four additional tube materials and to document the pitting resistance curves such that they may be used by heat exchanger designers and plant support personnel.

FY02 Deliverable:

An interim Technical Report documenting the pitting corrosion data developed in FY2002.

FY02 Estimated Cost: **$**175K

Total Estimated Cost: $325K

3-223 Revision to ASME Section XI Appendix G, RPV Pressure-Temperature Limits

Principle Objective: The objective of this effort is to completely revise the methodology in ASME Section XI Appendix G governing the development of reactor pressure vessel (RPV) pressure-temperature (P-T) limits during plant heat-up and cool down.

Need: The procedures outlined in Appendix G are based on fracture mechanics methods and constitute a flaw tolerance approach to vessel integrity where the margin on pressure, postulated flaw size, and material toughness ensure that the material in the reactor vessel maintains a consistent and acceptable tolerance for flaws that might exist in the RPV. Conservatism exists in these procedures such that severe restrictions in operating flexibility are being imposed on operating plants. This narrow P-T operating 'window' increases the risk of low temperature over pressurization events. Significant enhancements have been made in the fracture mechanics and NDE technologies embedded in the Appendix G methodology. A complete revision to the Appendix G methodology is necessary to take advantage of these enhancements.

Expected Duration: 3 years

Scope of Work:

FY2002 Scope of Work:

Results will be utilized from the ongoing EPRI MRP/NRC PTS Reevaluation Effort to begin addressing the technical issues associated with a revision to Appendix G:

Subtask 1. Establish New Reference Flaw Size
Results of ongoing efforts to develop a revised flaw distribution for use in PFM analyses, and recent enhancement of nondestructive evaluation (NDE) capabilities will be utilized to establish a revised reference flaw size for use in a future flaw tolerance procedure to be developed under the project.

Subtask 2. Develop Improved RPV Integrity Assessment Methodology
Efforts will be initiated to develop an improved RPV integrity assessment methodology for consideration in a revised Appendix G. This will include incorporation of the Master Curve fracture toughness methodology as well as consideration of cladding residual stresses, cladding/base-metal stress free temperature, and flaw stress intensity solutions.

Subtask 3. Modification of PFM Analysis Methodology
Efforts will be initiated to modify the PFM analysis methodology (ORNL FAVOR code) presently being developed in support of the EPRI MRP/NRC PTS Reevaluation Effort. Modification of the FAVOR code will be performed in order to support the technical basis for a revision to Appendix G.

Work Scope for Subsequent Periods:

Subtask 3. Modification of PFM Analysis Methodology (FY03)
Efforts will be completed regarding modification of the PFM analysis methodology.

Subtask 4. Develop Technical Basis for Appendix G Revision (FY03)
The results from Subtasks 1-3 will be utilized to develop a technical basis for a revision to ASME Section XI Appendix G.

Subtask 5. ASME Code Change (FY04)
While participation in appropriate ASME Code committees will continue throughout this project, a code change based on the developed technical basis will be presented to Section XI in FY04 for incorporation. This subtask supports the code activity through committee participation and supporting technical analyses as requested.

FY02 Deliverable:

Progress report for Subtasks 1-3.

FY02 Estimated Cost: DOE: $400K EPRI: $700K

Total Estimated Cost: DOE: $750K EPRI: TBD

3-224 Master Curve Fracture Toughness Implementation

Principle Objective: Incorporate the Master Curve fracture toughness methodology into codes, standards, and regulations governing reactor pressure vessel (RPV) integrity assessment

Need: The Master Curve fracture toughness approach is being pursued by industry as a more accurate, less conservative methodology for determination of RPV material reference temperature. The testing procedures have been standardized by ASTM but implementation of the methodology into plant operating criteria and regulations is necessary to realize benefit in RPV integrity assessment.

Expected Duration: 3 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

<u>Subtask 1. Margins Development</u>
Present RPV integrity-related regulations include application of a margin term to account for material variability and other uncertainties in the determination of the current Charpy-based RPV material reference temperature. Utilization of the Master Curve fracture toughness approach will reduce the amount of uncertainty that is required to be applied. This subtask will document the uncertainties to be considered and recommend appropriate margins associated with the Master Curve approach for plant operating criteria and regulatory consideration.

<u>Subtask 2. International Guidelines Development</u>
Efforts are underway to develop international guidelines regarding the application of the Master Curve approach to RPV integrity assessment. This subtask will provide for participation in the development of international guidelines to ensure consistency with ongoing Codes and Standards activities. Supporting testing activities are included under this task.

<u>Work Scope for Subsequent Periods</u>:

<u>Subtask 2. International Guidelines Development (FY03)</u>
The development of international guidelines governing application of the Master Curve approach to RPV integrity assessment will be completed and documented. The impact of these guidelines on domestic RPV integrity assessment will be evaluated.

<u>Subtask 3. Codes and Standards Implementation (FY03)</u>
Requirements for the testing of ferritic materials to determine the fracture toughness-based transition temperature through use of the Master Curve approach have been standardized by ASTM in Standard E1921-97. However, the operational benefits from this technology will not be fully realized until the Master Curve approach is completely integrated into existing plant operating criteria as an alternative to the Charpy-based approach. This will require an extensive revision to existing ASME Section III and Section XI. This subtask will pursue appropriate Code revisions and develop accompanying technical bases for Code body consideration.

Subtask 4. Integrated PWR Surveillance Program (FY03)
The ability to predict future embrittlement of RPV materials using the Master Curve approach will require an extensive database of irradiated surveillance materials testing using the Master Curve approach. Specific plant materials have already been tested under utility-directed programs. However, an integrated PWR surveillance program is necessary to coordinate activities between PWR owners, identify the critical materials to include in a Master Curve surveillance program and efficiently administer program activities to maximize the generation of relevant data. This subtask will develop and integrated PWR surveillance program in cooperation with NSSS vendors.

Subtask 5. Fracture Toughness Embrittlement Trend Curve (FY03 and FY04)
Results from Subtask 4, along with additional laboratory testing of archived, irradiated materials to be performed under this task, will be utilized to develop an embrittlement trend curve of fracture toughness data. The trend curve will incorporate mechanistic considerations and be statistically robust to allow prediction into the life extension time period.

FY02 Deliverables:

- Technical report documenting margins to be applied for application of Master Curve approach
- Progress report regarding international guidelines development.

FY02 Estimated Cost: DOE: $300K EPRI: $400K

Total Estimated Cost: DOE: $1,250K EPRI: TBD

3-225 Crack Propagation Study of PWR Core Internals using Small Specimen Designs and Test Techniques

Principle Objective: Using newly developed small-size bend specimens to optimize usage of neutron-irradiated materials, the cracking behavior of stainless steels will be determined under static and dynamic loading conditions relevant to the crack-tip stress-strain states and thermal conditions of PWR baffle bolts.

Need: Cracking of core internals has prompted increased surveillance and monitoring efforts at enormous cost, which will significantly increase as plants apply for re-licensing. Therefore, it is essential that irradiation- and environmentally-induced cracking of high-strength stainless steels be determined and subsequently related to material microstructure to minimize monitoring efforts. Additionally, limited neutron-irradiated materials are available for such testing, therefore, adopting newly-developed small bend specimen designs can yield significant cost savings. Furthermore, these bend specimens will provide meaningful cracking susceptibility data from sharp cracks under triaxial states of stress, which is not possible using other small specimen techniques (e.g., shear punch, automated-ball indentation, “mini”-tensile). For future testing and to an even greater benefit, it is possible to machine large quantities of small bend specimens directly from as-tested, full-size compact tension specimens (such as those for JOBB and CIR Programs) for direct comparison of crack growth rates from different test techniques.

Expected Duration: 4 years

Scope of Work:

FY2002 Scope of Work:

Materials from EBR-II surveillance irradiations (304, 347, 17-4PH, and possibly 304/308 welds irradiated to 15-20 dpa at 370°C) will be machined into miniature single-edge notched bend (SENB) specimens. Likewise, for baseline comparison to irradiated materials, non-irradiated sibling materials, which have been aged under comparable thermal conditions, will be machined into miniature SENB specimens and tested under static and dynamic loads for determination of crack growth rates as a function of stress intensity, fatigue cycle, and temperature. Corresponding microstructural and fractographic examination using optical and electron microscopies will be performed on thermally-aged materials.

Work Scope for Subsequent Periods:

FY 2003 Scope of Work

- Complete crack growth studies of thermally-aged materials.
- Perform crack growth studies of neutron-irradiated EBR-II materials under various bending deflection rate and low-cycle fatigue conditions at temperatures from 26-370°C, and
- Perform detailed microstructural and fractographic characterization of neutron-irradiated EBR-II materials using optical and electron microscopies.

FY 2004 Scope of Work
- Perform comparable crack growth studies of thermally-aged/non-irradiated and irradiated materials in simulated PWR coolant, and
- Continue detailed microstructural and fractographic characterization of neutron-irradiated EBR-II materials using optical and electron microscopies.

FY 2005 Scope of Work
- Complete testing and characterization.
- Complete final summary report.

FY02 Deliverables:

- Demonstrate capability to EDM-machine miniature SENB specimens from thermally-aged and irradiated materials.
- Demonstrate capability to test thermally-aged materials within a hotcell environment.
- Perform crack growth studies of thermally-aged stainless steels under various bending deflection rate and low-cycle fatigue conditions at 26-370°C.
- Perform detailed microstructural characterization of thermally-aged stainless steels using optical and electron microscopies.

FY02 Estimated Cost: $400K

Total Estimated Cost: $1,800K

3-226 Developing Ni-Alloy Welds Resistant to Stress Corrosion Cracking

Principal Objective: Assessment of the potential for weld-induced susceptibility to stress corrosion cracking in Alloy 152 welds and development of an optimized shielded-metal-arc (SMA) procedure to fabricate cracking-resistant Ni-alloy welds

Need: Recent studies have shown that cracking can occur in the weld regions of BWR core shrouds even in the absence of thermal sensitization (carbide precipitation on grain boundaries) and at fluence levels well below those believed to be necessary to produce susceptibility to IASCC in base metals. The cracking appears to be associated with contamination of the weld and heat-affected zone by oxygen, calcium, and fluorine introduced via the flux materials used in SMA or submerged-arc (SA) weld procedure. The observed increase in susceptibility may be due to a synergism among O, F, and Ca contamination which leads to F-catalyzed enhanced corrosion on grain boundaries (or interdendritic boundaries). Repairs for cracking such as those observed at V. C. Summer (RPV nozzle welded to primary piping) and Oconee (CRDM penetration nozzles welded to RPV) PWRs typically involve Alloy 152 welds using an SMA procedure. These replacement materials are inherently more resistant to cracking, but there is a potential for susceptibility to stress corrosion cracking due to the contamination that can be introduced by the SMA welding procedure. Although this problem can be avoided by using other procedures such as gas–tungsten–arc (GTA) welding, in many cases SMA procedures are much easier to perform and less costly, and an optimized SMA procedure which minimized the potential for contamination and cracking would have substantial benefits.

Expected Duration: 3 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

Alloy 152/Alloy 690 plate weldments will be produced by SMA procedures and the weld and heat-affected zones will be examined by advanced analytical methods such as SIMS that are sensitive to O, Ca, and F and can map the distribution of these elements. If contamination of the weld and heat–affected zones similar to that observed in the stainless steel core shroud welds is found, an optimized weld electrode coating, which minimizes contamination of the weld and heat-affected zones by O and Ca and eliminate contamination by F will be developed.

The Inconel 152 SMA weld electrode, described in Table 1, contains Ca- and F-containing compounds in its flux coating, i.e., $CaCO_3$ of 1-5 wt.% and Na_3AlF_6 of 5-10 wt.%. Because Inconel 152 welding electrode uses a coating with significant amount of volatile Ca- and F-containing compounds, contamination of weld and buffer zones by Ca and F is inevitable. In addition, significant O contamination from air and oxides that dissociate in arc is also inevitable. By developing and using alternative welding electrodes in which $CaCO_3$ and Na_3AlF_6 compounds are replaced with alternative compounds that do not contain Ca or F, the susceptibility of Inconel-buttered welds to stress corrosion cracking can be mitigated or virtually eliminated. Because we seek in this work to eliminate the sources of F and Ca that exacerbate the susceptibility to stress corrosion cracking, the functions provided by CaF_2, Na_3AlF_6, AlF_3, and $CaCO_3$ (i.e., the compounds that provide shielding gas and fluxing-cleaning agent) must be properly understood. CO_2 and CO gases, produced from decomposition of $CaCO_3$ and $(Ca,Mg)CO_3$ at high temperatures, are the primary components of the shielding gas. This function can be provided by $MgCO_3$ or $SrCO_3$, which do not contain Ca. Shielding ions of Ca^{++} and F^- are also produced during arc welding at high temperatures via dissociation of CaF_2. Then, Ca^{++} ions capture O rapidly from the air and F atoms are released in gaseous or plasma form. At the same time, parts of the Ca and F atoms dissolve in the weld and heat-affected zones. The most attractive alternative compound that plays the role of CaF_2 is YI_3, which undergoes a reaction similar to CaF_2, i.e.,

$$2CaF_2\,(s) + O_2\,(g) = 2CaO + 2F_2\,(g) \quad \text{vs.} \quad 2YI_3\,(s) + 3O_2\,(g) = 2YO_3 + 3I_2\,(g).$$

Because O stoichiometry in YO_3 is 1.5 times higher than that of CaO, more O atoms can be captured per mole of YO_3 than CaO. The melting temperature of YI_3 is lower than the melting temperature of CaF_2, i.e., 1004°C vs. 1418°C; therefore, dissociation of YI_3 will occur faster and at temperatures lower than that of CaF_2. When they are mixed, most halide compounds lower the melting temperature of oxides. Therefore, as with CaF_2, YI_3 is expected to perform well as a fluxing agent. This will complement the function of $MgCO_3$ and Na_2SiO_9 which do not contain F or Ca but also act as fluxing agents. In FY2002, sets of modified Inconel 152 welding electrodes will be developed to optimize the resistance to stress corrosion cracking of Ni alloy welds. The proposed modifications are limited to the composition of flux materials contained in the electrode coating. The modified electrodes will be produced in cooperation with one or more commercial suppliers of welding electrodes.

Table 1. Composition (in wt.%) of various types of Inconel welding electrode used in shielded-metal-arc welding of dissimilar metals

Type [a]	$CaCO_3$	CaF_2	Cr	Co	Fe	Mn	Mn_3O_4	Mo	Ni	Nb	SiO_2	Na_3AlF_6	$Na_2Si_4O_9$	$SrCO_3$	Ti	TiO_2	W
112	5-10	-	15-40	-	1-5	-	-	5-10	40-70	1-5	1-5	5-10	1-5	-	-	1-5	-
117	5-10	-	15-40	5-10	1-5	0.5-2	-	5-10	40-70	-	0.5-2	5-10	1-5	-	-	1-5	-
122	5-10	1-5	15-40	-	1-5	-	-	10-30	40-70	-	0.1-1	5-10	1-5	-	-	1-5	1-5
152	1-5	-	15-30	-	5-10	1-5	-	-	40-70	1-5	0.1-1	5-10	1-5	1-5	-	1-5	-
182	5-10	-	15-30	-	5-10	1-5	1-5	-	40-70	1-5	0.1-1	1-10	1-5	-	1-5	1-5	-

[a]Inconel ID number of welding electrodes.

<u>Work Scope for Subsequent Periods</u>:

Prepare welds using the optimized weld electrodes. Conduct bend-beam SCC tests on the Alloy 152/Alloy 690 welds in simulated PWR water.

FY02 Deliverables:

Six types of modified Inconel 152 SMA weld electrodes.

FY02 Estimated Cost: $300K

Total Estimated Cost: $800K

3-228 Microwave Sensor for Non-contact Condition Monitoring of Containment

Principle Objective: This applied research endeavor pertains to the application of microwave (MW) sensing techniques for remote condition monitoring of concrete-based structures and in particular the containment vessel of a nuclear power plant (NPP). Some areas of immediate concern include detection of voids, cracks, and rebar corrosion in reinforced concrete that can lead to loss of strength and quick deterioration of service-degraded structures. The primary objective is to devise a remotely operated microwave reflectometer array for wide-area inspection of life-limiting defects in containment structures such as cracks, voids, and corrosion/breakage of rebars.

Need: Safety concerns for maintenance and life extension of the aging nuclear industry infrastructure and the ever increasing demand to improve both productivity and performance have led to a greater demand for improved nondestructive testing (NDT) methods. Remote condition monitoring of the containment vessel is one such area of application. Cost effective aging management based on degradation susceptibility requires implementation of non-invasive sensing techniques for routine monitoring of irradiated structures. Feedback from routine in-service inspections can provide guidance to establish mitigation strategies for more appropriate aging management of nuclear power plants. Microwave techniques offer many advantages over conventional methods for inspecting materials with low conductivity such as masonry. Such sensors are intrinsically non-contact (i.e., require no coupling medium) and allow for high-speed interrogation of large areas. Recent advances in monolithic integrated circuit (MMIC) technology combined with the availability of portable high-speed computers for data acquisition and processing have made possible integration of cost-effective remote sensors for various NPP industry inspection applications.

Expected Duration: 3 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

Rapid non-contact measurements, for the determination of susceptible material condition, are necessary for cost-effective implementation of more frequent inspections that could ultimately lead to improved performance and operational safety. Anisotropic nature of masonry such as concrete (e.g., composition, density variations, presence of rebars, porosity, etc.) and their poor conductivity has limited the usefulness of many conventional NDT methods for their inspection. Microwave techniques, on the other hand, can offer significant advantages for inspecting materials with low electrical conductivity. The large bandwidth associated with microwaves allows for high-resolution and selective measurement of constituents in a composite material. Also, the wave nature of EM radiation at these frequencies would permit implementation of real-time imaging techniques. MW sensors can be operated in a non-contact mode and at relatively large standoff distances which is a basic requirement for the rapid interrogation of large structures. This also renders such sensors particularly suitable for operation in harsh environments (e.g., presence of heat, stress, vibration, and airborne particulates).

Scope of the proposed work for the initial phase of studies on utilization of MW probing technology for non-contact inspection of containment structure will encompass three fundamental tasks, namely,

1. Employment of analytical and numerical models to study the process of propagation and scattering of EM waves in composite dielectric media at MW frequencies.
2. Implementation of small-scale model experiments using laboratory-based instrumentation to test the concept.
3. Compilation and analyses of test-bed data.

Theoretical studies will aid toward efficient optimization of measurement parameters. For experimental studies various test specimens will be subjected to near- and far-field microwave radiation. Scattered fields in both regions will be measured and analyzed. Coherent and incoherent field pattern data will be processed to gain a better understanding of the interaction of the radiating field with the medium under examination. Subsequently, the effect of simulated defects such as inclusions, voids, cracks, excess moisture content, and corroded reinforcements will be fully studied. Attempts will be made during this stage of work to best simulate the conditions encountered in the field. Experimental efforts will primarily emphasize on detection and characterization of defected reinforcements, aggregate size distribution (in-homogeneity), voids, cracks, and possibly the stress and strength properties of reinforced concrete using wide band MW measurements. The study of the changes in the detected fields will yield information about the defect and energy interaction and eventual characterization of potential flaws. A correlation of microwave signature to critical material integrity parameters will be sought. Utilization of various signal polarizations and a measurement confidence analysis based on the experimental results will be developed toward the end of the initial phase of this work. These studies will be designed to devise an optimum measurement scheme for characterizing different types of irregularities.

<u>Work Scope for Subsequent Periods</u>:
Upon successful completion of the initial phase a prototype MW remote inspection system potentially consisting of transmit/receive unit, electronics, and data acquisition and analysis hardware/software will be assembled. This effort will encompass:

- Design and assembly of microwave subsystems.
- Construction of matched directional antennas for optimal coupling.
- Design and assembly of electronic components; implementation of appropriate data analysis and display routine.
- Test the concepts and systems in the field.

It is envisioned that the final stage of this work will involve system demonstration at a selected test site that will be carried out in close cooperation with a designated NPP industry partner.

FY02 Deliverables:

The initial phase of this applied research endeavor will demonstrate the feasibility of the proposed MW sensor by generating ground truth data using laboratory-based instrumentation and simulated small-scale model experiments. This study will produce quantitative information to resolve the primary design issues needed for integration of a prototype in the second phase of this work. A detailed report on the results of these investigations will be submitted at the end of the first year.

FY02 Estimated Cost: $250K
Total Estimated Cost: $1,000K

3-229 Non-Destructive Inspection of Inaccessible Portions of Nuclear Power Plant (NPP) Metallic Pressure Boundaries

Principle Objective: Guided ultrasonic wave modes (Long Range Lamb Waves) will be used for detecting, quantifying and locating general corrosion and pitting in the embedded portions of nuclear power plant (NPP) metal pressure boundary so that its structural and leak-tight integrity can be assessed.

Need: Embedded portions of light water reactor metal containment are susceptible to general corrosion and pitting by groundwater permeating through the basemat concrete or by leaking fluid collected at the concrete-metal interface where a gap contains filler material such as sand that can retain fluid. The embedded metal portion is also susceptible to corrosion if the sealant at the metal-concrete interface is broken. The corrosion challenges the structural integrity of the containment and, if through-wall, can provide a leak path to the outside environment. Corrosion of embedded steel has occurred at the concrete-metal interface in pressurized water reactor (PWR) ice-condenser containment and boiling water reactor (BWR) Mark I drywell bases where wall thicknesses were determined to be below the design limits. Conventional ultrasonic inspection techniques can not be used to detect corrosion in the embedded portion of the metal containment and, therefore, advanced nondestructive inspection techniques are required.

Expected Duration: 3 years

Scope of Work:

FY2002 Scope of Work:

Multi-mode guided plate waves (Lamb waves) will be used in a through-transmission operation using small diameter transducers that require minimal access to the metallic surfaces. The plate wave modes travel long distances, depending on the frequency and mode characteristics of the wave, and follow the contour of the structure in which it is propagating. The use of plate wave modes is potentially a very attractive solution since they can be excited at one point on the structure, propagated over considerable distances and received at a remote point on the structure. The received signal contains information about the integrity of a portion of the structure that lies between the transmitting and receiving transducers. This also has the advantage that the wave can propagate through structures (e.g., the embedded portion of the BWR Mark I containment shell), which are inaccessible to conventional inspection techniques. It has been demonstrated that nondispersive modes, which propagate long distances and are sensitive to wall thinning, cracks, and pits that may be present in the material being examined. Using Lamb wave techniques long-range measurements of 100 meters or more have been demonstrated.

During the first year, a feasibility study will be performed to evaluate the Lamb-wave technique for detecting, locating, and quantifying corrosion damages in thin carbon steel plates. Measurements will be made in samples with and without mechanically-simulated corrosion damage and with and without concrete embedment. The study will consist of both theoretical modeling that can predict various modes of plate waves and their propagation characteristics and laboratory tests from which Lamb-wave parameters will be correlated with defects due to corrosion damages. The tests will also determine the selections of transducer and operating modes. Both pitch-catch and pulse- echo modes of operation will be evaluated; the former will be used to determine the uniformity of the plate thickness and the latter to locate the damage.

A transient finite element model will be developed to study Lamb wave interaction with simulated corrosion damage in the embedded plate samples. This model will be bench marked with the measurements. Then, the model will be used to assess the effect of embedded plate size and thickness, and extent of corrosion damage and its location on the sensitivity (dispersion) of the selected modes.

<u>Work Scope for Subsequent Periods:</u>

<u>FY2003 Scope of Work:</u>

Shell sample and Test Stand Fabrication:
A one-tenth scale shell sample will be obtained and subjected to machining and chemical attack to simulate typical pitting and corrosion damage. Tests will be conducted to evaluate the proposed Lamb wave inspection technique.

Further Analytical Development:
Transient finite element model to study Lamb wave interaction with simulated corrosion damage in the embedded carbon steel shell sample will be developed and investigated with the guided plate waves.

Lamb Wave Technique Development:
Sensor design, data collection, the data interpretation and quantification algorithm, and demonstration of technology on the scaled test samples will be undertaken. This data will be used as a calibration data set for quantitative estimation of damage. New signal regeneration techniques will be explored for improvement of signal-to-noise ratio (SNR) and for discrimination of the damage mechanism which has been detected (i.e. Corrosion vs. Pitting). Ultrasonic data will be acquired on plates containing areas with/without simulated damage (a) before embedment, and subsequently (b) after embedment in sand and concrete.

Evaluation of Reliability and Accuracy:
Key parameters such as false alarm rate, probability of detection, transducer and technique specifications, limitations, will all be studied and estimated. In order to verify the approach, it is proposed to consider developing a qualitative correlation between the inspection results using the plate waves and the half-cell potential measurements. The half-cell potential measurement method can potentially detect corrosion damage in the embedded portion of the containment vessel.

<u>FY2004 Scope of Work:</u>

Prototype Instrument Development & Technology Transfer Initiative:
A portable PC- based prototype instrument will be designed with switchable control for the generation and recording of the ultrasonic data using the Lamb wave techniques. Extensive interaction with the industry collaborator (to be identified later) will take place to finalize the

design and prototype instrument development and to establish the mechanism for the transfer of the technology to the industry partner.

Final Report and Demonstration:
A final report will be prepared documenting all work efforts undertaken and the results. The inspection technique will be demonstrated in a field test, site for which will be selected based on plant availability and maintenance schedule.

FY02 Deliverables:

Feasibility Study Report:
A report describing the results of the feasibility of using the guided plate waves for detecting, quantifying and locating general corrosion and pitting damage in carbon steel plate embedded in concrete will be prepared. This report will include an assessment of the analytical model developed to study Lamb wave interaction with simulated corrosion damage in the plate samples.

FY02 Estimated Cost: $300K

Total Estimated Cost: $1,200K

Generation Optimization Projects

5-103 Dry Storage of Spent Fuel with Burnup in Excess of 45 GWd/MTU

Principle Objective: To develop the technical basis and confirmatory experimental data for regulatory acceptance of practical approaches for storing spent fuel with burnup in excess of 45GWd/MTU under dry, inert atmosphere.

Need: The current industry trend toward longer fuel cycles drives reactor operators toward higher burnup fuel designs. By the end of 1998, approximately 5,000 PWR assemblies and 1,000 BWR bundles, with assembly-average burnups in excess of 45 GWd/MTU, had been discharged to wet storage at the U.S. reactor sites. Starting in 2001 for PWRs, and just a few years later for BWRs, the majority of assemblies to be discharged in any given year will be characterized by assembly-average burnup greater than 45 GWd/MTU. By 2007, over 90% of assemblies to be discharged will have burnups greater than 45 GWd/MTU.

Presently, the Nuclear Regulatory Commission's Standard Review Plans (NUREG-1536 and -1567) do not address dry storage of spent fuel having burnups in excess of 45 GWd/MTU. In Interim Staff Guidance 11 (ISG-11), Rev.1, the NRC Staff has proposed permitting storage and transportation of high-burnup fuel assemblies under conditions which are considered penalizing (costly) and less-than-practical by industry.

The technical basis and associated methodologies to be used for licensing dry storage systems for spent fuel with burnup in excess of 45 GWd/MTU are presently the subject of discussions between NRC's Spent Fuel Project Office and Industry (NEI/EPRI/Storage System Designers). The industry-proposed technical basis relies on analytical considerations supported by experimental information derived from accelerated testing on cladding tube segments. However, experimental validation of the behavior of spent fuel rods and assemblies under prototypical conditions will eventually be needed for confirming the validity of the technical basis as well as for supporting interim regulatory acceptance.

Failure to resolve this issue in a timely manner (by end of 2004) would result in severe economic penalties (upward to ~$5M/reactor-cycle in storage costs) and in operational limitations deriving from spent fuel pool constraints (unquantified costs).

Expected Duration:	Task 1:	2 years	
	Task 2:	Phase I – 1 year;	Phase II – To Be Defined

Scope of Work:

<u>FY2002 Scope of Work</u>:

The scope of work of this activity is:
Task 1: Development of a Technical Basis for Dry Storage of High-Burnup Fuel

Work will continue in support of two EPRI reports already submitted to the NRC and in further developing the technical basis and supporting methodology for performing creep analysis of spent fuel cladding under dry storage conditions.

Task 2: Demonstration of High-Burnup Fuel in Dry Cask Storage, Phase I: Recommended Approach

At least, two options for a demonstration program will be considered. The first option consists of storing pre-characterized, high-burnup spent fuel assemblies (potentially with different cladding types) in a dual-purpose (storage and transportation) system designed for high burnup fuel; appropriate modifications for monitoring fuel cladding integrity and measuring temperatures inside the cask will also be considered. The second option, similar to the apparatus used for the Fuel Assembly Internal Temperature Measurement (FAITM) test facility, consists of an experimental storage system in which a pre-characterized, instrumented spent fuel assembly is stored under prototypical dry storage conditions. The practicality (costs, transportation requirements, etc.) of these options will be evaluated; a recommended approach will be developed.

Work Scope for Subsequent Periods:

Task 1:
Development of a Technical Basis for Dry Storage of High-Burnup Fuel (Funded by Industry)

Regulatory acceptance of the technical basis and supporting methodologies for cask storage of high-burnup spent fuel will be pursued through NEI.

Task 2:
Demonstration of High-Burnup Fuel in Dry Cask Storage, Phase II
Phase II will be proposed as a new project based on the results of Phase I. At the present time, it is anticipated that Phase II will consist of the following:

Assuming regulatory acceptance of the technical basis developed under Task 1, a confirmatory program directed at providing confirmatory data on heat transfer, fuel integrity, cladding deformation, and retrievability will be conducted. The scope of the program will include pre-characterization of the assemblies before dry storage, real-time monitoring of fuel integrity and cladding temperatures during dry storage, and non-destructive (visual, dimensional, …) as well as destructive (hydrogen content and distribution, mechanical properties, …) examinations after ~5 years of dry storage.

FY02 Deliverables:

- Technical Report on "Creep Modeling and Analysis -- Methodology for Spent Fuel Dry Storage" (in support of Task 1).

- Feasibility Study for conducting the Demonstration Program, including recommended experimental set-up, institutional considerations, detailed estimates of costs, and options (in support of Task 2).

FY02 Estimated Cost:	$800K	Task 1:	EPRI: $600K	
		Task 2:	DOE: $150K	EPRI: $50K

Total Estimated Cost: $800K

5-106 Low Power and Shutdown Probabilistic Risk Assessment (PRA) Qualitative Research Pilot Plant Proof of Concept / Low Power and Shutdown Human Reliability

Principle Objective: This project has two objectives: 1) Perform a proof of concept evaluation of the low power and shutdown qualitative criteria developed in 2001 NEPO 5-106 project. 2) Understand nature and causes of human errors during LPSD operations.

Need: 1) A current year NEPO project (5-106) will develop qualitative criteria for low power and shutdown conditions based on existing quantitative shutdown PRA efforts. The American Nuclear Society in 2001 is developing a qualitative probabilistic risk assessment standard for shutdown operations utilizing this research. This project will perform a proof of concept of the 5-106 research. 2) Traditional PRA human reliability methods and analysis are crafted around full power operations. These methods and analysis may not be fully applicable to LPSD conditions. This project will develop comparable analytical techniques for LPSD operations with special attention to organizational factors

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

Most utilities have developed qualitative low power and shutdown (LPSD) analysis based on NUMARC 91-06. In addition, several utilities have or are developing quantitative full scope LPSD PRA models. Because it these full scope LPSD PRA models are difficult to develop and utilize, most utilities rely on qualitative methods to ensure safe operation during shutdown. Hence the work in NEPO project 5-106 is currently being performed to develop the qualitative criteria. Once the criteria are developed, they should be benchmarked. Therefore, this project will work with a utility to perform a proof of concept of the research results developed in NEPO project 5-106 by one of the following methods:

1) Utilizing an existing plant qualitative analysis, compare this analysis against the qualitative criteria developed in NEPO 5-106. This will determine inconsistencies between the existing plant analysis and the research. As appropriate and to the extent possible with the project resources modify the existing model to adhere to the research criteria.

2) Work with a utility to incorporate into a new LPSD qualitative analysis the research results. Project resources will be used to supplement utility resources in the analysis development.

3) Collect and collate LPSD human reliability event experience into a database. Data will be gathered from industry sources such as LERs, EPRI reports, etc., as appropriate.

Work Scope for Subsequent Periods:

Review the data gathered in Task 3 above to determine if shutdown experience is sufficiently different from full experience warranting changes to existing full power HRA models and methods. If required develop recommendations in HRA methods for use in shutdown PRAs.

FY02 Deliverables:

- Pilot plant implementation report documenting the lessons learned from the implementation of the standard.
- Interim report documenting LPSD database of human errors.

FY03 Deliverables:

Final report documenting LPSD findings from data analysis, and if appropriate, recommended changes to HRA modeling techniques.

FY02 Estimated Cost: DOE: $270K EPRI: $235K

Total Estimated Cost: DOE: $585K EPRI: $550K

5-110 Guidelines for Hybrid Control Rooms

Principle Objective: To provide guidance for definition, design, implementation, operation, and maintenance of, as well as training for, hybrid control rooms including human related issues. (Hybrid describes control rooms containing mixed analog and digital technology.) The hybrid control room resulting from modernization of plant systems and associated human-system interfaces is a major focus for this project. The guidance will support, in addition to meeting NRC safety requirements and the plant's operational requirements, improved cost-effective plant and human performance and reduced likelihood of human errors, resulting in improved plant safety, availability, reliability, and cost-effective operation. In addition, there will be work to develop technical bases to support the development of design guidance in new critical advanced areas related to control room modernization that have not be addressed for nuclear power plants by other organizations in the past, such as changes in process automation and computerized operator support systems.

Need: Many I&C systems are of analog design, and contain components that are or soon will be obsolete. In many instances, analog replacements are no longer available. Plants are finding it necessary to procure digital-based designs as replacements. The move from analog to digital is accelerating as plants age and as more plants receive approval for license renewal. With the long life of most existing plants (due to young age of the plant and/or license renewal), modernization of I&C equipment is no longer optional as most of the equipment will not support the plant until the end of the plant's life. A major issue related to this modernization is the resulting control room which will be hybrid in nature at least for several refueling cycles, if not until the end of plant life.

Careful digital I&C system design and integration into control rooms can reduce the likelihood of human errors as well as improve human performance. Conversely, human performance weaknesses may occur without proper attention to definition and design as shown in an NRC study of five events involving digital technology that found human performance weaknesses in several areas. The NRC has developed several NUREGs and related NUREG/CRs but they are only intended to guide NRC review to ensure a plant meets safety requirements. They are not designed to provide guidance needed for defining, designing, implementing, operating and maintaining hybrid control rooms including human related issues. For utilities and designers there needs to be extensive expansion past what is given in the NRC documents. Some examples of issues to be addressed are:

- Defining what should the final control room look like and what are its capabilities at the end of a multi-cycle modernization program, also provide what capabilities and appearance during the various steps of the modernization program leading to the final state..
- Determine how to train operators for hybrid control rooms that change substantially at least every refueling outage.
- Designing systems and procedures to handle system faults and abnormal conditions including loss of the computers in the control room under a different control room paradigm.
- Determining when and how to replace a fixed position analog control or display with a “soft” digital unit (selectable when needed but not always shown).
- Evaluating need for, value of, and effective design of systems to provide increased automation, alarm suppression/filtering, computerized procedures, decision aids, integrated displays/controls, etc.
- Designing window-based presentations to support plant stakeholders such as operations, engineering, and maintenance.

There are several advanced technology aspects of the modernization programs for which the technical basis to support guidance development is not readily available. This project will develop the technical basis in these areas for use to develop the needed design guidance.

Due to the urgency of the results from this project in many areas, a prioritization of issues will be developed based on utility input on needs. The project will produce several interim guidelines with technical bases so that the guidance will be available in a timely manner. The guidelines will utilize input and advise from appropriate parts of the industry including equipment suppliers, control room operators, design, engineering, maintenance, and licensing. This input will be obtained through working group discussion as well as site visits.

Expected Duration: 3 years

Scope of Work:

FY2002 Scope of Work:

- Continue to define the contents of the guidance document and supporting materials based on utilities' and designers' needs. Although most of this definition will be done in the first year, additional issues will come up from additional interactions with the utilities and designers.
- Continuing development of technically valid and defensible technical bases for the guidance for hybrid control rooms. The technical bases are being developed based on standards, regulations, guidelines, current practices, research results, and other appropriate resources both domestic and international. Nuclear industry resources and resources from other areas such as aerospace, military, universities, process industry, etc., will be used as appropriate. Current and modern state-of-the-art practices are to be included. Advanced plants and advanced plant designs will be considered both for an identification of the areas to be addressed as well as for guidance. The regulatory positions of the NRC must be well understood and factored into the technical bases developed, as well as in the hybrid control guidance itself.
- Continue to develop guidance that will be user friendly to make it usable and useful for designers and utility engineers who are not human factors engineers. This guidance will be value-based to make sure that it can be applied cost-effectively. Risk-informed considerations will be taken into account, as appropriate. The guidance will be valid and technically defensible and will be practical to use. The guidance will be able to be implemented using currently available technology and equipment as well as anticipating future equipment. The guidance will include checklists as much as possible to facilitate the use of the guidance by designers and utility engineers. The range of applicability of the guidance will be identified.
- Test and evaluation efforts to address the usability, acceptability, completeness, benefit etc., of the guidance with utility and designer users will be done. Utility and designer personnel will be used to support this test and evaluation task.

- Continue development of supporting materials to facilitate guidance document use, and to upgrade and maintain the documents. A training program will be created for users of the guidance documents. Guidance regarding procedure upgrades will be prepared. Modifications to human-system interfaces and the control room will require changes to existing and possibly creation of new operations and maintenance procedures.
- A needs analysis will be conducted to define the most important advanced aspects of digital I&C and control room modernization for which technical basis information is not currently available, but which will be needed. Project personnel will work with industry experts to define and prioritized these topical areas.
- Information on each of the selected topics will be collected from a wide range of sources, including: published literature, vendor design documentation, lessons learned from with utilities and input from subject matter experts. The information search will include both the U.S. and foreign nuclear industries. Modernization programs have been more extensive in many foreign plants, thus they are an important source of information. Where needed, information will also be obtained from other industries, such as non-commercial nuclear facilities, petrochemical facilities, and aerospace systems.
- The results of the technical basis development will be integrated into topical reports. It is anticipated for planning purposes that five such topical reports will be developed.

Work Scope for Subsequent Periods:

- Complete development of technically valid and defensible technical bases for the guidance for hybrid control rooms including human related issues.
- Complete development of guidance for hybrid control rooms including human related issues.
- Complete test and evaluation of guidelines.
- Complete development of supporting materials to facilitate guidance document use, including training program, and to upgrade and maintain the documents. Complete guidance regarding procedure upgrades.
- Completion of the development of the technical basis for advanced topics.
- Completion of the development of the technical basis topical reports for advanced topics.

FY02 Deliverables:

Deliverables will include interim guidelines, technical bases for the interim guidelines, and results from evaluating the interim guidelines. Simplified versions of the guidelines in the form of checklists will also be produced. Supporting materials including training course to support application of the guidance document, and plan to update the guidance document will be delivered. Due to the urgency of the results of this project, intermediate products based on an urgency prioritization by the utility working group for this project, will be delivered throughout the length of the project.

FY02 Estimated Cost: DOE: $370K EPRI: $640K

Total Estimated Cost: DOE: $1,005K EPRI: $1,880K

5-113 On-Line Monitoring of Non-Redundant Sensors for Improved Performance

Principle Objective: Demonstrate on-line monitoring technology in operating nuclear plants for a variety of systems and applications. Verify that on-line monitoring using a signal validation method such as the Multivariate State Estimation Technique (MSET) is capable of identifying instrument drift or failure under a variety of conditions.

Need: Safe and reliable operation of nuclear power plants depends on the ability to monitor and control plant operations. On-line monitoring can directly support this need, while also optimizing the on-site maintenance and testing activities. On-line monitoring as a performance assessment tool is needed for the following types of applications:

- Assessment of instrument and equipment health
- Determination of calibration
- Long-term equipment and system performance trending
- Enhanced troubleshooting capabilities

Expected Duration: 2 years.

Scope of Work:

FY2002 Scope of Work:

EPRI has formed the Instrument Monitoring and Calibration (IMC) Users Group to provide instrumentation and control (I&C) services to EPRI members in technology transfer, training, and implementation of key IMC products. During FY2001, the IMC Users Group initiated the transition from product demonstration to product implementation. The scope of work described here directly supports the implementation of on-line monitoring at each participating nuclear plant by optimizing the various applications of on-line monitoring. The following tasks are planned:

Optimization of Sensor Fault Detection Methods. Statistical tools have been developed to detect the onset of signal drift or failure. Several tools are available to improve fault detection capability; however, the applicability of these methods to nuclear power plant systems have not been evaluated fully. Evaluate missed alarm/failed alarm settings, correlation limits, signal filters, and signal comparison methods to optimize the fault detection capability for the selected systems.

Optimization of Model Training Strategies: The number of sensors in a given model, the signal variation for each sensor over the range of interest, the quality of the training set, and the time interval between training data sampling affect the subsequent ability of the software to identify faults for various combinations and types of signal drift or failure. Using data provided by the project participants, evaluate methods to optimize the training strategies considering the above variables.

Uncertainty Analysis

Perform an uncertainty analysis to quantify the effect of numerous known variables. The uncertainty analysis should consider the following: number of sensors; number of observations in training set; range of values in training set; linear correlation coefficient value (or other measure) between sensors over some reasonable range; quality of training set; accuracy of sensor outputs; missed alarm/false alarm settings; signal conditioning applied during the analysis, such as regularization, or other means, to smooth/reduce noise content; and effect on model of data outside training range.

Development of Models to Optimize Plant Efficiency and Thermal Performance

Develop models for the purpose of evaluating plant efficiency and thermal performance. The intent of these models is to identify efficiency losses as they occur while the nuclear power plant is operating at its normal steady-state power level.

Verification and Validation (V&V) of Software, Including Simulated Test Sets

Perform a V&V of the software used for on-line monitoring to document its accuracy and reliability. The purpose of this V&V effort is to develop the basis documentation necessary to allow the on-line monitoring tool to be used at operating nuclear plants, as well as to develop simulated test cases for testing the software's features.

In support of the above tasks, project participants will provide data as well as initial models for evaluation. The data provided by a single plant will generally cover some portion of an operating cycle, typically including a minimum of 4,000,000 individual measurements.

Work Scope for Subsequent Periods:

As the project finishes its specified effort for FY2001 and FY2002, additional tasks are planned with regard to modeling and analysis of systems that experience considerable process or seasonal variations. Fault detection methods will be tested for these more complex applications.

FY2002 Deliverables:

- Technical report documenting the uncertainty analysis and the simulations used for the analysis.
- Technical report describing the fault detection and training strategies that provide most benefit to typical nuclear power plant systems.
- Technical report describing the results of applying on-line monitoring for the purpose of optimizing plant efficiency and thermal performance.
- Documented V&V of on-line monitoring method, included simulated test sets.

FY2002 Estimated Cost: $260K

Total Estimated Cost: $510K

5-117 R&D Needs to Address Potential Nuclear Plant Vulnerabilities Arising From Transmission Grid Voltage Inadequacies

Principle Objective: Task 2 System Prototype / Demonstration
Task 3 Advanced Calculational Methods and Display Concepts

Need: These task is part of the work required to assess the vulnerability of nuclear power plants due to various transmission grid problems, such as voltage sag, congestion, or disruptions. As the electric utility industry now undergoes its most fundamental restructuring in a century, the challenge of maintaining grid reliability is growing. With increases in inter-regional bulk power transfers and accelerating diversity of transactions among parties, the electric power grid is being used in ways for which it was not originally designed. Grid congestion and unusual power flows are increasing in a way that degrade the voltage support needed by nuclear plants per a plant's technical specifications --- and increases the probability that a nuclear plant forced to trip off line for licensing or other reasons will further destabilize the grid.

Expected Duration: 3 years

Scope of Work:

FY 2002
Task 1 (to be completed under 2001 funding) provides for the creation of a software link between the risk monitors (e.g. EOOS, Safety Monitor) used in nuclear plants and a wide area grid transmission monitor (called Real-time Security Data Display or RSDD). Task 2 will be for the prototype field testing of this communications link. Task 3 will produce advanced computational techniques and improved display systems for the plant and transmission dispatch centers.

FY02 Deliverables:

- Task 2 - A software module(s) conforming to the functional specifications developed under Task 1 in 2001 and a report detailing the experience using the prototype.
- Task 3 - Models (and functional specifications) that anticipate instabilities in the local area of electric transmission networks from information obtained from wide(r) area sensing systems and a report on new display systems developed.

FY02 Estimated Cost: DOE: $150K EPRI: $150K

Total Estimated Cost: DOE: $440K EPRI: $440K

5-201 LOCA Qualifiable Digital Transmitter Based on Fiber Optics

Principle Objective: To develop a fiber optic transmitter capable of withstanding 200 megarads and accident steam conditions. The transmitter will be used with digital electronics located in a benign environment.

Need: The integrated circuits used in digital transmitters cannot withstand the radiation conditions from a Loss of Coolant Accident. Therefore, to allow digital technology to be used for safety transmitters located in containment, the digital electronics must be located in a benign environment. A fiber optic based transmitter allows this to be done and reduces the qualification of the electronics to seismic and electromagnetic compatibility testing and validation and verification of digital soft and firmware.

Expected Duration: 2 years

Scope of Work:

FY2002 Scope of Work:

European researchers have identified fiber optic cable types and a fiber optic element capable of withstanding 200 megarads of irradiation. This research will develop a differential pressure transmitter based on the cable and element. The ability of the transmitter to measure pressure with adequate accuracy and repeatability will be verified under normal. The prototype will be exposed to extremes of accident environments to determine if susceptibilities exist.

Work Scope for Subsequent Periods:

A full prototype will be developed and subjected to design and environmental qualification.

FY02 Deliverable:

Results of testing of the prototype will be recorded in a report including accuracy, repeatability, and ability to withstand extremes.

FY02 Estimated Cost: DOE: $250K; EPRI: $125K

Total Estimated Cost: DOE: $400K; EPRI: $275K

5-202 Guidelines for the Monitoring of Aging of Nuclear Power Plant I&C System Electronic Boards

Principle Objectives:
1. Collect data on electronic board and component failures and aging in nuclear power plants
2. Identify and describe aging and failure mechanisms for electronic boards and components in nuclear power plants
3. Develop a guide for monitoring and forecasting aging of electronic boards and components in nuclear power plants

Need: The nuclear power industry is currently facing increasing aging and obsolescence issues with original equipment installed for instrumentation, control, and safety system applications. One area that needs attention is the aging of electronic boards and components used in instrumentation and control (I&C) systems in nuclear power plants. To manage these electronic parts better, it is necessary to understand the aging and failure mechanisms that occur with them. For utilities, the aging of electronic boards and components has to be considered when coping with several concerns. One concern is the desire to optimize the cost of maintenance programs which requires answers to questions such as when should a preventative replacement of a part be done; and is the observed failure a generic phenomena or an isolated case? Another concern is to increase the operational lifetime of the systems which requires answers to questions such as what is the best way to handle the boards; is the environment of the board harmful to it and its components; and what actions can be taken to prevent the board and components from aging? A third concern is for input into deciding whether a system needs to be modernized which requires answers to a question such as how much longer can the system be safely and economically used? This evaluation can then be combined with other considerations such as obsolescence, performance, and desired new functionality to come up with a decision.

Work is needed to identify and study the aging and failure mechanisms of electronic boards and components to better understand the consequences for the electronic parts in nuclear power plants. Guidance on monitoring and on forecasting aging would be extremely beneficial to the plants. This project is designed to do this.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

- In order to better understand the current situation and to develop guidance on the monitoring of aging of I&C electronic boards and components in nuclear power plants, field data on failures and aging needs to be collected. The first step will be to define, with participation of several utilities, the list of parts to be considered in this project. As additional information is found, this list may be revised. The project team will contact as many utilities as possible to get their aging and failure data on these parts.
- Data will be collected from the various utilities and will be analyzed. The data is expected to be in the form of answers to interviews, pictures of boards and components, general information and comments, reports, failure rates, and failure analysis reports. This data and the results of the analysis

of the data will be documented in a report. The report will include list of the electronic board and components failures reported by the utilities, including root causes, analyses, etc., and any information that would help to achieve a diagnosis when a new failure occurs.

- Information will be collected from literature on every aging mechanism for each family of parts of interest. The information will be extracted with at least the description of the mechanism, the root cause of the aging/failure mechanism, its effects, some basic recommendations for prevention if available, and the frequency of the mechanism. This will be documented.
- Information will be collected from literature on the methods which are available to anticipate or monitor the aging mechanisms for electronic boards and components. This information will be analyzed, and in some cases, the methods will be assessed with real electronic boards and parts taken from utilities to see if they can be adapted to the utilities' environment where special constraints have to be taken into account (e.g., the systems may be monitored but must be available and functioning, the population of the parts is rather small, etc.). New methods will be developed to anticipate or monitor aging mechanisms and assess them.
- A guide will be developed to tell which methods can and should be used to monitor and forecast the aging of electronic boards and components.

<u>Work Scope for Subsequent Periods:</u>

Work will continue on the aging and failure mechanisms of electronic boards and components. A report on aging and failure mechanisms of electronic boards and components in nuclear power plant I&C systems and guidance for the monitoring of aging of I&C electronic boards and components in nuclear power plants will be developed.

FY02 Deliverables:

- Final report on collected field data on electronic part failures and aging in nuclear power plant I&C systems.
- Interim report on aging and failure mechanisms of electronic boards and components in nuclear power plant I&C systems.
- Interim guide for the monitoring of aging of I&C electronic and components in nuclear power plants.

FY02 Estimated Cost: DOE: $85K EPRI: $85K

Total Estimated Cost: DOE: $195K EPRI: $195K

5-204 Qualification of Commercial Digital Components for Replacement of Obsolete Equipment in Nuclear Safety Systems

Principle Objective: The principle objectives of this project are to make needed digital components available for use in replacing aging and obsolete equipment and to provide examples and lessons learned that demonstrate a predictable, proven process for qualifying commercial digital equipment for use in nuclear safety applications. The project will perform and document tests and evaluations required to qualify selected commercially available components, with particular emphasis on the most problematic issues, including electromagnetic compatibility, software quality, and failure analysis.

Need: To ensure continued high reliability and availability, nuclear plants need to replace many aging and obsolete instrumentation and control (I&C) components for which qualified replacements do not currently exist. Qualification of equipment for nuclear safety applications involves various types of testing and evaluation to ensure that it is of adequate quality for its intended use – to ensure that it will perform its safety function, even when subjected to various postulated adverse environmental and operational conditions. The components of interest for this project are self-contained devices that are relatively inexpensive, ranging in cost from a few hundred to several thousand dollars, and that are used in many applications in the plants. For these devices the qualification costs far exceed the component costs. Examples of such components include controllers, pressure transmitters, recorders, time delay relays, temperature sensors, digital panel meters, power supplies, pressure-to-current signal converters, and pressure switches.

Operating nuclear plants for the most part still use I&C equipment that was designed and built decades ago under specialized nuclear quality assurance programs. In the intervening years, most industries have transitioned from the analog technology used in the nuclear plants to digital technology, and the vendors have updated their product lines accordingly. When nuclear plant design and construction stagnated in the 70's, the demand for nuclear qualified equipment declined to the point where few vendors maintained their nuclear quality assurance programs or their qualified products. Processes were developed to qualify many types of commercial components for use in nuclear safety applications, but this did not extend to software-based I&C equipment. The result is that most operating nuclear plants are using a great deals of obsolete I&C equipment for which qualified replacements simply don't exist.

The impact of equipment obsolescence is amplified by the fact that most nuclear plants are now planning to extend their operating licenses by twenty years. Just a few years ago, most plants were planning to nurse the old equipment along until the plant license expired, and then shut down. Now, with renewed emphasis on extending the usefulness of existing power generation assets, most nuclear plants are looking ahead to decades of operation, and continued maintenance of the obsolete equipment is no longer a practical option. Declining reliability is just the tip of the iceberg. Repair often is difficult or impossible without reverse engineering or manufacturing custom parts, and the needed expertise on both the technology and the power plant systems is gradually being lost as the workforce ages and retires. To meet reliability and availability goals, the plants have no option but to replace their obsolete control equipment over the next several years.

The transition from analog to digital technology has further exacerbated the problem. While digital technology brings many performance and reliability advantages, it also raises new technical and regulatory concerns. The process for qualifying analog equipment for nuclear safety applications is well established, but the corresponding process for digital equipment, and especially for commercial grade digital equipment, is still evolving. At this point, qualification of a digital device can be far more complex and expensive than for other types of hardware, and once undertaken, there is no guarantee that it will be successful. For many relatively inexpensive components and devices, neither the manufacturer nor individual utilities can practically bear the risks associated with such an effort. The process needs to be improved to the point where utilities and suppliers can engage in a qualification effort with predictable costs and schedule, and with high confidence in its ultimate success in regard to both technical and regulatory challenges.

Various different types of components must be addressed. Nuclear plant I&C systems necessarily encompass a broad range of functionality, and the components of interest in this project vary correspondingly, with widely differing design characteristics and needs from the qualification perspective. Specific attributes such as software complexity, testability, operating history, and failure modes, all of which can vary widely, are key to designing and implementing a component-specific qualification strategy that demonstrates adequate overall quality and dependability. Treating several candidates of different types will generate the needed range of strategies and examples and lessons learned that can be applied in future qualifications.

If this project is not implemented, nuclear utilities will continue to struggle with their obsolete equipment. Needed digital replacements will ultimately be qualified, but at a much slower pace and a higher per-item cost. Lower reliability will be tolerated, with correspondingly increased maintenance costs and failure rates, and corresponding threats to plant availability. Realization of the potential performance and reliability advantages of digital technology will be delayed several years at least. It is likely that in some cases, plants will be forced to replace failed obsolete equipment on an emergency basis, either by extending outages or during forced outages. In either case, for one such instance at one plant, the cost in replacement power and lost availability would likely exceed the cost of this entire project.

Also, if this project is not implemented, the process for qualifying commercial digital equipment will continue to be unpredictable and unevenly practiced across the industry. Stabilizing the process is important, because the transition from analog to digital technology will necessarily continue, with increasing use of commercial equipment. Also, the digital equipment becomes obsolete more rapidly than its analog predecessors, so the qualification of a set of digital replacements for obsolete components is not a one-time event. A steady supply of new digital equipment that is qualified for nuclear safety applications will be needed, which means an established, predictable qualification process will be needed.

This project will also allow the nuclear plants to approach the equipment suppliers as an industry, rather than on a plant-by-plant basis. This is important, because some of the qualification activities require vendor participation, including detailed discussion with key technical staff. For most of the suppliers of components that this project would address, the nuclear industry represents at most a few percent of their market, and the special evaluations, tests, and documentation that the nuclear industry requires tend to make them reluctant to participate. On this project we can approach the vendors as an industry, with the assurance that the results will be made available accordingly, and that individual utilities will not have to return on a case-by-case basis. EPRI can also serve as a focal point for industry notifications, should problems be discovered that need to be communicated to the plants. The group approach will conserve resources for all parties involved, and will also reduce regulatory risk associated with using a qualified component.

The most important impact of this project is that it will help ensure high reliability and availability for the remaining lives of operating nuclear plants. To accomplish this, the project will provide several qualified components, but it will also provide a proven, predictable process tested on several different types of components. A side benefit will be a new pool of knowledgeable equipment suppliers, consultants and utility engineers who can support future qualifications of digital components.

Expected Duration: 2.5 to 5 years (depending on the funding level)

Scope of Work:

FY2002 Scope of Work:

The first step will be to augment and prioritize the list of candidate components, using input from utility participants in the program, and to develop a qualification strategy for each candidate. The strategy will be tailored to the specific component. It will consider such issues as the breadth of applicability of the device, willingness of the manufacturer to participate in the evaluations, availability of software documentation, existing data on seismic and electromagnetic testing, options for supporting the qualification throughout its useful life, along with qualification costs, and likelihood of success.

Also in the first year, qualification activities will be started on at least one high priority candidate. The work will involve planning, performing, and documenting evaluations and testing needed for nuclear safety qualification. The project activities will be based on the most up to date guidance and standards on the use of digital equipment in safety-critical applications, as well as existing processes for qualification and commercial grade dedication for nuclear safety applications. Issues addressed will include: seismic ruggedness; electromagnetic compatibility; operating history; behavior under faulted or abnormal conditions; and supplier practices and controls, including manufacturing, software development, configuration management, and problem reporting. It is anticipated that by the end of the first year, the tests and evaluations of the first components will be largely complete, and the documentation will be in preparation.

Work Scope for Subsequent Periods:

Qualification efforts on additional components will be conducted, subject to funding limitations. After the up front priority/strategy task, the project is modular and can be managed on a component-by-component basis, with appropriate consideration of the variability in qualification costs from one component to the next. By using multiple contractors to treat several components in parallel, the project can be accelerated (up to an estimated six components at a time). Accordingly, estimated costs are shown below for two approaches which each address a total of twelve components - a ‘minimal’ project approach (one component the first year, two the second, and three in the third, fourth and fifth years) and an ‘optimal’ project approach (four components the first year, six in the second, and two in the third).

FY02 Deliverable:
Report describing candidate components, including recommended priorities, qualification strategies, and estimated qualification costs.

FY02 Estimated Cost:	\$550 K	(DOE: \$341K; EPRI: \$275 K) (Minimal feasible approach)
	\$1,600 K	(Optimal approach over 2.5 years)
Total Estimated Cost:	\$4,400 K	(DOE: \$2,266; EPRI: \$2,200) (Twelve components)

5-206 Development of a Safety-Critical Architecture for Embedded Applications Implemented Using Commercial-Off-The-Shelf (COTS) Hardware and Software

Principal Objective: The objective of the proposed research is the development of a safety-critical architecture which allows the use of Commercial-Off-The-Shelf (COTS) hardware and software in the design and implementation of safety-critical digital Instrumentation and Control (I&C) systems for plant digital upgrades.

Need: The primary issue that will be addressed during this project is the need to design and quantify the safety of safety-critical digital processor-based Instrumentation and Control (I&C) systems implemented using Commercial Off The Shelf (COTS) hardware and software. Currently, these safety-critical systems are designed and implemented using custom hardware and software, and the safety quantification of such systems can only be performed experimentally, which is extremely resource intensive. In addition, the quantification typically has to be performed for each application since the safety properties of the system are closely tied to the implementation.

The primary benefit of this project would be that it provides a technique for designing safety-critical digital I&C systems that allows the use of COTS hardware and software, thus making it a more cost effective solution than designing custom hardware and software. Also, it would provide a safety-critical architecture whose safety can be quantified analytically (based on numerical assurance techniques), and thus does not require the tremendous amount of resources that are typically needed to experimentally quantify the safety of a system. In fact, the quantification process only needs to be performed once because the safety properties of the system are independent of the implementation and transparent to the application developers.

Expected Duration: 3 years

Scope of Work:

This project will continue to develop safety-critical design technology that has been produced at the University of Virginia over the past 10 years. The proposed project will construct an experimental prototype designed using the developed safety-critical architecture that will be used to implement a selected safety-critical I&C application from the nuclear industry. An experimental test-bed will be developed to demonstrate the safety provided by the safety-critical architecture by performing fault injection experiments on the selected application.

<u>FY2002 Scope of Work</u>

Task 1: Error models for combined hardware/software systems - Safety is characterized using a three-universe model that defines the physical, information, and external universes. Faults occur in the physical universe and are defects or imperfections in a physical entity. Errors occur in the information universe and are corruption to units of information as a result of a fault. Finally, failures occur in the external universe and represent non-performance or substandard performance of some expected function. Existing state-of-the-art safety-critical design approaches depend on a detailed understanding of the anticipated fault set which implies specific

knowledge of the system's physical hardware and software. The result is a tight linkage between the safety properties of a system and the implementation-specific hardware and software. In Task 1, error models of combined hardware/software systems, rather than models of faults, must be developed. The outcome of Task 1 will be the error models which can be used in the design and analysis of dependable systems using COTS modules to allow the safety properties to be independent of the application.

Task 2: Algorithm-Based Fault Tolerance (ABFT) techniques - ABFT techniques form a class of numerical assurance approaches which attempt to develop a system's safety properties by focusing on the algorithm to be executed rather than the hardware/software on which the algorithm is implemented. In Task 2 an ABFT technique will be developed which can concurrently verify the correct execution of an algorithm such as one commonly used in embedded control and signal processing applications. The proposed approach has several unique features. The proposed ABFT technique confirms all aspects of the algorithm including: (1) use of the correct operands, (2) performance of the correct operations, (3) correct performance of the operations, (4) performance of the correct sequence of operations, (5) correctness of the information used throughout the algorithm, and (6) correctness of the time associated with all data. The approach is applicable to any algorithm which can be created as a series of basic operations. The basic operations include arithmetic, logical, control flow, and relational functions, and form a computationally complete set.

Work Scope for Subsequent Periods

Task 3: Design of a proof-of-concept prototype using the developed safety-critical architecture that will be used to implement a selected safety-critical I&C application from the nuclear industry.

Task 4: Development of an experimental test-bed to demonstrate the safety provided by the safety-critical architecture by performing fault injection experiments on the selected application.

Task 5: Modeling and analysis techniques for ABFT systems - Task 5 will develop the modeling and analysis techniques needed to calculate the probability of an undetected error. The proposed ABFT technique depends on compression approaches to simplify the error detection process. The negative impact of compression is an aliasing effect, which can make some errors undetectable. There are two limitations in the state-of-the-art techniques for the analysis of compression properties: (1) the inability to properly handle correlated errors, and (2) the inability to predict the transient properties of the probability of detection. In Task 5, analysis techniques will be developed which overcome both of the existing limitations. In addition, the techniques will be extended beyond the analysis of a single node to the analysis of a distributed system containing many such interacting nodes.

FY02 Deliverables

- Deliverable 1: A report will be produced describing the error models which can be used in the design and analysis of safety-critical systems using COTS modules to allow the safety properties to be independent of the application (corresponds to Task 1).

- Deliverable 2: A report will be produced describing the Algorithm-based Fault Tolerance (ABFT) techniques which can concurrently verify the correct execution of an algorithm such as one commonly used in embedded control applications. (corresponds to Task 2).

FY02 Estimated Cost:	DOE: $125K	EPRI: $125K
Total Estimated Cost:	DOE: $462K	EPRI: $463K

5-213 Using Deterministic Analysis Tools and Statistic Combinations of Uncertainties (SCU) Methodology to Support Instrument Calibration

Principal Objective: Verify that the MSET technology can adequately represent plant instruments over a wide range of conditions. This will be accomplished by applying EPRI's SCU methodology to plant instrument calibration and verify results against existing database of over 300,000 plant measurements and using deterministic analysis to supplement the plant data.

Need: On-line monitoring for instrument calibration reduction will improve plant reliability and safety, reduce O&M costs and reduce radiation exposure. EPRI submitted a topical report TR-104965 to the NRC that endorsed methodologies for instrument calibration reduction. The methodology included technology developed by Argonne National Laboratory called Multivariate State Estimation Technology (MSET). This technology is also used for signal validation and has been demonstrated at various plants. The NRC has concerns associated with this methodology as it is basically a "curve fit" of data and may break down in an unanticipated manner.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work:</u>

The first year is anticipated to be focused on using deterministic codes and the EPRI SCU Methodology to be model the necessary plant components and instrument trains. Demonstrating that the model is valid and that a response surfaces that can be used. A limited set of instrument and plants will be used to demonstrate feasibility.

<u>Work Scope for Subsequent Periods:</u>

The methodology developed earlier will be extensively validated using the existing plant database. The various response surfaces need to support the overall licensing effort will be developed and documented.

FY02 Deliverable:
A technical report documenting the feasibility study performed and how this can be used to gain confidence in the earlier work.

FY02 Estimated Cost: $200K

Total Estimated Cost: $500K

5-214 LOCA Initiating Event Frequency Derivation

Principle Objective: Develop Loss of Coolant Accident (LOCA) pipe break initiating event frequency data from recently developed pipe break database.

Need: Probabilistic risk assessments have a portion of the identified risk attributed to pipe break LOCAs. With the development of new industry databases on pipe failures, a new source of more accurate initiating event frequency is available.

Expected Duration: 1 year.

Scope of Work:

FY2002 Scope of Work:

Several databases on nuclear power plant pipe failures have been created recently. An industry standard database is being compiled from previous source by the OECD NEA. It is proposed that this database or a comparable database be used to derive small and medium pipe break initiating event frequencies.

Updated generic initiating event frequencies will be developed for appropriate initiating event categories, systems and reactor type. Using Bayesian techniques working on a prior and the new data, a new set of IEF distributions will be generated including mean, 5^{th} and 95^{th} percentile values for each U.S. reactor type and vendor based on the available databases by performing a Bayesian update of the prior.

- Develop appropriate binning of initiators as necessary.
- Capture appropriate point estimate prior distributions.
- Develop new generic industry distributions by updating the prior distributions with events from approximately 1990 – 2000 (inclusive) using Bayesian techniques.
- Document results in final report.

Work Scope for Subsequent Periods:

No follow on work is anticipated.

FY02 Deliverable:

Final report documenting initiating event frequencies with distributions with full description of methodology used.

FY02 Estimated Cost: DOE: $150K EPRI: $100K

Total Estimated Cost: DOE: $150K EPRI: $100K

5-218 Nuclear Plant Safety Risk Management

Principle Objective: Risk-informed regulation is an objective of nuclear regulators and nuclear plant owner/operators. In order for risk-informed regulations to proceed and be effective, owner/operators must be able to demonstrate the capability and effectiveness of safety risk management. EPRI Nuclear Sector has a strategic objective to define the risk management process (both configuration control and performance monitoring) that plants must employ to support risk-informed and performance-based regulations and operations. This project will define the elements of safety risk management, perform case studies on plants with effective risk management, identify beneficial generic improvements, and communicate these results to stakeholders.

Need: Implementation of risk-informed regulation reform is proceeding slowly. Acceptance criteria for changes to account for "safety assurance" and for "uncertainty" are too conservatism. The concept of acceptable levels of risk and performance monitoring to validate risk assumptions are not truly accepted by all stakeholders. Until these issues are resolved, cost-effective risk-informed operation is unlikely. More importantly without these resolutions, risk informed regulations may not be implemented to a significant degree and the associated improvements in plant safety will not be attained.

Expected Duration: 2 years

Note: The defined scope of this project is expected to be complete in one year. However, one critical element to assure effectiveness of safety risk management is effective collection and use of information. Specifying the information management requirements and performance of a pilot application of the results are expected to be a logical follow-on to this project for another year. Achieving the primary objectives of this work scope is not contingent on continuation of the project beyond the first year.

Scope of Work: EPRI has been a leader in developing risk management tools and processes for nuclear plants, including ORAM-Sentinel, EOOS, R&R Workstation, and the Outage and On-line Risk Management Forum. These tools and methods are widely used. EPRI has been instrumental in supporting Maintenance Rule (especially paragraph (a)(4)), Regulatory Oversight Program monitoring, risk-informed Tech Specs, and other programs that are elements of risk management. Furthermore, EPRI is investigating ways to identify and prevent critical latent human errors and leading-indicators of risk-important conditions.

Progressing separately, these programs significantly reduce safety risk and financial risk. Considered as an integrated process, they contribute to a safety culture that has greater near-term and long-term advantages:

- Objective evidence that risk is maintained at acceptable levels
- Objective evidence that risk informed burden reduction has the expected safety and cost effects

- Leading indicators used to find and reverse adverse trends and degraded conditions prior to an event
- Tools to reduce unnecessary conservatism and margin in operational decisions
- Risk models to support improvements in organizational factors

Specific tasks in the 2002 work scope will include:

1. Create a conceptual model of a nuclear plant risk management program. The model will identify important functions, processes, and their relationships; and it will be general enough to accommodate a variety of specific tools and methods.
2. Identify existing plants having many of the attributes of the conceptual model in place.
3. Map the risk management processes at one or two of these plants onto the conceptual model to produce a "Risk Management Overlay" for the plant.
4. Evaluate the effectiveness of the risk management processes, and define changes to enhance the process. Changes can include new or better metrics, enhanced analysis tools, awareness training, organizational or program adjustments. Opportunities for generic products to improve risk management will be identified for potential follow on work.
5. Write a report intended for owner/operator management and regulatory policy decision-makers. The objective of the report will be to communicate the power of risk management to facilitate a change to a risk-informed and performance-based safety culture and to provide a risk-management assessment and enhancement process for use at plants.

FY02 Deliverable:

The final report described in Task 5 above will be the 2002 deliverable.

FY02 Estimated Cost: DOE: $200K EPRI: $200K

Total Estimated Cost: DOE: $475K EPRI: $475K

5-220 Guidelines for Wireless Technologies in Nuclear Power Plants

Principle Objective: Develop technical guidance to support the deployment of wireless technology applications in nuclear power plants without compromising safe and reliable operation.

Need: Improve human and plant performance through technology integration using wireless applications. Benefits include:

- Reduced O&M costs
- Improved plant monitoring
- Reduced radiological exposure

Expected Duration: 2 years.

Scope of Work:

EPRI has formed the Wireless Working Group to assess the needs of industry guidance to support the application of wireless solutions in nuclear power plant applications. These applications include video, voice and data networking; paging; cellular and personal digital assistants (PDAs). The impact of high frequency wireless devices on existing plant equipment is unknown. The purpose of this effort is to assess the risk and provide technical guidance to support introducing wireless devices in nuclear power plants.

FY2002 Scope of Work:

- Risk Analysis. The study will assess operating experience with wireless devices in power plant environments as well as in other industries (pulp and paper, chemical, etc.).Plant events including trips and transients caused by wireless devices will be evaluated and their impacts quantified to a level of risk. A detailed OE analysis will be documented in a report which will assist plant personnel in understanding potential risks.

Guidelines for Evaluating Wireless Technologies for Power Plant Applications. The purpose of this research is to provide comprehensive guidance to support evaluation of wireless technologies for nuclear power plants. This effort will focus on the development of rules structure to support the deployment of wireless devices in the plant without compromising continuous, safe and reliable operation. The research will include:

- Wireless protocols for transmission of voice/data/video information
- Radio-frequency interference (RFI) impacts on existing plant equipment
- Conformance with industry standards
- Data security
- Data quality

Work Scope for Subsequent Periods:

Additional tasks are planned to upgrade the guidelines and risk analysis documents based on technology advancements, utility experience, and regulatory feedback.

FY2002 Deliverables:

- Technical report documenting the risk analysis based on operating experience
- Progress report on the status of the development of the guidelines.

FY2002 Estimated Cost:	DOE: $75K	EPRI: $75K
Total Estimated Cost:	DOE: $150K	EPRI: $150K

5-221 Development of New Algorithms for use in Digital Protection and Monitoring Systems

Principal Objective: The purpose of this program is to develop advanced software for use in digitally based protection and monitoring systems; and evaluate the impact of such software in terms of enhanced margin to safety, and potential for thermal margin gain. Such software has the potential to dramatically enhance the economic benefit of upgrades to digitally based systems. The program focuses on Westinghouse and older CE PWR designs. There may be as many as 40 to 60 plants that could benefit from these advancements.

The goal of the program will be to identify and develop potential software that provides up to an additional 10% margin with respect to various safety limits. A portion of this margin can be used for Power Uprates, thus significantly enhancing the economic benefits of digital upgrades.

Need: The management of the degradation of analog instrumentation and control components in many operating plants is reaching a critical point. Many plants are considering or are re-embarking on upgrades (modernization) utilizing digital-based systems to take advantage of a commercial base of vendors, improved resistance to calibration drift, and increased capability for self-diagnosis, data handling, and easy maintenance; as well as improved availability and reliability of these systems.

However, these digital systems also provide a significantly enhanced calculation capability over their analog counterparts, whose calculation capability was limited by circuitry and cabinet space. The potential of this new capability for enhanced safety and thermal margin gain has not been evaluated, as almost all new digitally based systems use the same original logic that existed based on limited calculation capability of analog circuitry.

Expected Duration: 3 years

Scope of Work:

The program will target the Westinghouse and CE design PWRs that utilize analog protection and monitoring systems. It will use the existing experience gained from the application of this concept in Core Protection Calculators (CPC), Core Operating Limit Supervisory System (COLSS) and Best Estimate Analyzer for Core Operations-Nuclear (BEACON); as well as the existing simulation tools which are used in safety analysis of the operating Westinghouse and CE designed PWRs. It will examine the NSSS behavior during the events which yield the most severe challenge to key safety limits. The goal is to develop advanced software for use in the protection and monitoring systems that would increase the safety margin and/or potential thermal margin gain through enhanced system response to NSSS transients. Such software can be designed to use only the measurement signals for parameters that are currently available for processing in the unit. Compatibility with existing safety analysis methods will be considered in evaluation of the benefit of the new algorithms.

FY2002 Scope of Work:

Investigation and development of software algorithms.

Work Scope for Subsequent Periods:

Quantification of margin gains for one demonstration plant (year 2).
Preparing two Functional Design Specifications, for Reactor Protection and Reactor Monitoring System (year 2).
Installation of the software on demonstration platforms (year 3).
Preparation of a draft licensing topical report for presentation to the NRC (year 3).

FY02 Deliverable:

Report describing the proposed software and qualitative evaluation of their potential for margin gain.

FY02 Estimated Cost:	DOE: $250K	EPRI: $250K
Total Estimated Cost:	DOE: $1,250K	EPRI: $1,250K

5-222 Developing An Optimized Procedure for Preparation of Spent-Fuel Prior to Dry Storage

Principle Objective: Develop an optimized procedure for initial drying and transfer of spent fuel assembly from storage pool to dry-storage cask to minimize hydride redistribution and reorientation in fuel cladding

Need: Stress-rupture and creep properties of medium- and high-burnup spent fuel cladding are strongly influenced by the size, distribution, number density, and orientation of hydrides. Hydride structure can be significantly modified during the initial process of vacuum drying and transfer of spent-fuel assembly from the storage pool to dry-storage casks. If the degree of hydride redistribution and reorientation is excessive, mechanical properties of the cladding are degraded significantly and the margin for survival under long-term creep is significantly reduced. To optimize the performance of spent fuel and to ensure a sufficient margin of cladding structural integrity during dry storage, it is desirable to minimize modification of the hydride structure by optimizing the drying and transfer procedures.

Expected Duration: 3 years

Scope of Work:

FY2002 Scope of Work:

Zircaloy-4 PWR fuel cladding contains up to ≈800 wppm hydrogen after high burnup (≈62 GWd/t). In as-spent condition and during pool storage (at ≈80°C), the hydride structure in the fuel cladding is well preserved. That is, mostly circumferential hydrides are confined in the "hydride rim" near the metal-oxide boundary, and the midwall and the inner-diameter (ID) side of the cladding remain relatively free of hydrides, thereby retaining a significant level of ductility and fracture toughness. This characteristic hydride structure of PWR spent-fuel cladding is a result of several factors, i.e., the texture of the cladding, the radial temperature gradient produced during normal operation, and the stress gradient produced under the high coolant pressure. However, results of recent experiments in CEA, IPSN, EdF, and Studsvik have shown that profound redistribution of hydrides occurs when high-burnup Zircaloy-4 cladding is exposed to 350-450°C for several hours. Under such conditions small hydrides precipitate during subsequent cooling in high-density out of dissolved hydrogen solutes (some hydrogen atoms are quenched-in during fuel discharge, some are from partial dissolution of the large circumferential hydrides. This type of hydrides are distributed in more or less random directions (i.e., in tangential, radial, and mixed directions), and the modified hydride structure leads to significant degradation of the mechanical properties of the cladding.

For most current designs of dry storage cask and the vacuum drying procedure, spent fuel cladding is exposed to 350-510°C for 10-50 hours during the initial drying and transfer period. Therefore, significant hydride redistribution is inevitable during this important period. It is not clear if significant level of hydride reorientation (i.e., formation of large radial hydrides) will or will not occur during the same period. However, during exposure to 310-380°C for 9-14 h and subsequent power-pulse experiments in IPSN, considerable hydride reorientation has been observed in local regions of high-burnup PWR fuel cladding. This has been attributed to localized residual stress due to rod bowing and PCMI that is sufficiently high to trigger hydride reorientation in local regions. Considering these findings, it is desirable to limit the peak cladding temperature to less than about ≈320°C to minimize general hydride redistribution and localized hydride reorientation, and therefore, to avoid undesirable degradation of mechanical properties. This can be achieved via an optimized drying cycle and assembly transfer procedure, e.g., by continued cooling with a flowing gas and subsequent sealing the cask with pressurized helium. The resultant effect will be drastically improved margin of survival under long-term creep conditions.

In the initial phase in FY2002, various drying cycles will be explored and analyses will be performed to identify the most promising candidate drying cycles and cask-sealing procedures, including drying and sealing in a helium-circulating container. Based on the results of the analysis, a small-scale mockup facility will be constructed that will be used to test the efficiency of the candidate drying cycles.

<u>Work Scope for Subsequent Periods:</u>

The hydride structure characteristic of Zircaloy-4 high-burnup spent fuel cladding will be produced in laboratory in oxygen-charged-and-homogenized unirradiated cladding in simulation of the hydride rim and irradiation-induced hardening of spent fuel cladding. These specimens will be exposed to the optimized drying and sealing procedure identified in FY 2002. Absence of redistributed or reoriented hydrides will be confirmed from a set of pressurized time-temperature heating and cooling experiments on mockup fuel assembly. From these experiments, the maximum allowed temperature that does not induce hydride redistribution and reorientation will be determined. An optimized drying cycle and sealing procedure will be recommended.

FY02 Deliverables:

Various drying cycles will be explored and analyses will be performed to identify the most promising drying cycle and cask-sealing procedure, including drying and sealing in a helium-circulating container. Based on the results of the analysis, a small-scale mockup facility will be constructed that will be used to test the efficiency of the candidate drying cycles.

FY02 Estimated Cost: $300K

Total Estimated Cost: $900K

5-227 Improved Temperature Measurements at U.S. Nuclear Power Plants

Principal Objective: To develop and demonstrate a more accurate measurement for temperature using laser-activated thermoluminescent phosphors

Need: Temperature is a primary parameter in power plant thermal efficiency. Conventional temperature sensors suffer from a variety of limitations such as drift, limited accuracy, EMI and radiation sensitivity, slow response time, and need for isolation from the process fluid. Fiber-optic temperature sensors have the potential for overcoming all of these limitations. Plant thermal power directly corresponds to the temperature rise of the primary coolant. A one percent too large differential temperature reading reduces plant power production by one percent. (EPRI has reported that at one plant a 1% temperature measurement error leads to a 3.7% decrease in turbine efficiency if the sensor reads high and loss of component life if it reads low. Resistance thermometer and thermocouple drift of a few percent is not uncommon in normal operation. Assuming a 3¢/kW-hr price for electricity, a one percent loss in generation for a 1000 MWe plant corresponds to a loss of $2.6 million per year. In addition to immediate operational savings, improved temperature measurement has the potential for increasing the maximum allowed plant thermal power. By decreasing the uncertainty in the peak process temperature, the required margin between maximum allowed operational conditions and equipment failure could be reduced.

A particularly applicable optical temperature measurement system is based on photoluminescent decay time. The basic measurement technique involves optically exciting a phosphor located at the tip of an optical fiber and monitoring the emitted fluorescence transmitted back down the same optical fiber. Since the lifetime of an excited atomic state is a function of temperature, the rate of decay of the fluorescence indicates the phosphor temperature. Moreover, since phosphor thermometry relies upon the decay time of an excited phosphor state, it is independent of signal strength and thus independent of attenuation in the light path, excitation source strength fluctuation, mechanical vibrations, and detector sensitivity drift. In addition, since the fluorescence process only relies on inner electron transitions and short range lattice order, it is resistant to crystal degradation and potentially will not require recalibration (degradation will manifest itself in a reduction of overall signal strength rather than as a change in time signature). With proper component specification, it may be possible to perform temperature measurements throughout nuclear power plants (including in-core) with this technique.

The phosphor thermometer developed for this project will be both mechanically and chemically durable, immune from EMI/RFI—will not present a shorting hazard, and will be highly stable to calibration drift. If the project is successful, the thermometer will decrease process temperature measurement uncertainties by more than an order of magnitude at higher temperatures and virtually eliminate the need for periodic thermometer recalibration.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>

- Develop small optoelectronics package to measure fluorescence decay rate of phosphors.
- Test chromium doped YAG phosphor for radiation response
- Determine if there is a lifetime shift due to radiation
- Monitor to see if excessive radiation darkening of optical fibers used for connection to measurement system

Work Scope for Subsequent Periods:

- Install in plant and verify performance
- Design and prototype second generation of opto-electronics to make optimally small

FY2002 Deliverable:

Report on design and analysis of first of a kind measurement system with improved accuracy and greater stability than existing instrumentation.

FY2002 Estimated Cost: DOE: $200K EPRI: $200K

Total Estimated Cost: DOE: $375K EPRI: $375K

5-232 Nano-scale Microstructural Analysis of PWR and BWR Cladding at Burnups exceeding 45 GWd/MTU: Correlation to Physical Properties

Principle Objective: Use advanced transmission electron microscopy (TEM) techniques to characterize neutron- and corrosion-induced defects in fuel cladding materials in order to correlate the observed microstructure to mechanical and oxidation properties of high burnup fuel cladding measured under conditions relevant to reactor transients and dry cask storage.

Need: In an effort to improve fuel management and LWR efficiency, increases in fuel burnup and coolant temperatures have been implemented. However, it is not yet known how these increases would impact cladding performance under severe mechanical and thermal loadings of design basis accidents and dry cask storage transients. Cladding irradiated to fuel burnups >45 GWd/MTU may become sufficiently embrittled and fail under severe loading resulting in significant fuel dispersal to the surrounding environment. Therefore, detailed microstructural characterization, coupled with corresponding macroscopic physical properties measured from mechanical and oxidation tests, are needed for a comprehensive understanding of high burnup cladding performance. In doing so, utilities and regulators may predict cladding behavior under specific transient conditions as discharge burnups approach licensing limits without the added cost of detailed post-irradiation examination and testing.

Expected Duration: 4 years

Scope of Work:

<u>FY2002 Scope of Work:</u>

A detailed microstructural characterization of LWR Zircaloy cladding will be initiated. Characterization will include a quantitative determination of dislocation structure (density, Burgers vector, and sub-grain, channel formation, and/or twinning behavior resulting from reactor-induced exposure), hydride and other second-phase precipitates (composition, orientation relationship, structure, size, and distribution), and reactor-induced oxidation (phase boundary structure, defect/grain boundary segregation, and bulk oxygen distribution). The data will be correlated to the mechanical and oxidation behavior of cladding materials from on-going experimental testing programs in the United States, France, Japan, and the Russian Federation. In FY02, characterization of the following cladding materials will be initiated:

- Recrystallized-annealed (RA) Zircaloy-2 irradiated to approximately 1.1×10^{22} n/cm^2 (E>0.1MeV) with peak fuel burnup of 62 GWd/MTU,
- Cold-worked, stress-relieved (CWSR) low-Sn Zircaloy-4 irradiated to approximately 1.3×10^{22} n/cm^2 (E>0.1MeV) with peak fuel burnup of 70 GWd/MTU
- Non-irradiated RA Zircaloy-2 and CWSR low-Sn Zircaloy-4 (control materials).

Work Scope for Subsequent Periods:

FY 2003 Scope of Work
Complete characterization study of unirradiated and as-irradiated cladding.
Partial characterization of unirradiated and irradiated cladding subjected to oxidation and mechanical testing relevant to LOCA and dry cask storage environments.

- Oxidation analysis (phase boundary structure, defect/grain boundary, segregation, and bulk oxygen distribution)
- 2nd-phase precipitate analysis (composition, orientation relationship,structure, size, and distribution)
- Deformation-induced dislocation analysis (density, Burgers vector, and sub-grain, channel formation, and/or twinning behavior resulting from reactor-induced exposure).

FY 2004 Scope of Work
Continue FY03 work.

- Obtain advanced cladding materials (Zirlo, M5, and/or E110) irradiated to high fuel burnup.
- Perform TEM image simulations of the damaged defect structure in selected.
- Cladding materials. Perform TEM data analysis regarding the physical properties of tested cladding materials.

FY 2005 Scope of Work
Complete FY03 work to include characterization of advanced cladding materials. Complete all data analysis and write summary report.

FY02 Deliverables:

- Quantitative determination of cold work-induced dislocation structure, 2nd-phase Precipitates, and oxygen distribution in nonirradiated sibling materials.
- Fabricate TEM specimens from irradiated cladding materials.
- Initial characterization of irradiated materials to quantitatively determine dislocation structure, hydrides and other 2nd-phase precipitates, and oxidation effects resulting from reactor exposure to high fuel burnup.

FY02 Estimated Cost: $250K

Total Estimated Cost: $1,050K

5-233 Cladding Oxide Spallation Mechanisms

Principal Objective: To determine the mechanisms that can lead to oxide spallation in high-duty fuel at high burnups and to determine whether oxide on advanced cladding materials is subject to spallation

Need: Spallation of a layer of cladding oxide leads to the formation of localized cold spots on a fuel rod. Hydrogen formed within the cladding as a result of the oxidation process, tends to migrate to such cold spots and eventually form hydride blisters. Such localized hydride blisters can have an adverse impact on cladding mechanical properties causing the cladding to become brittle. Oxide spallation has been observed on some PWR fuel with Zircaloy-4 cladding when such fuel is exposed to high burnups under high-duty operating conditions. Although oxide spallation has not yet been observed on the more advanced claddings (such as Zirlo or M-5), the mechanisms leading to spallation are not understood and there is a possibility that this phenomenon manifests itself in a different manner in such advanced cladding materials.
Fuel with spalling oxide may be subject to more restrictive licensing criteria for operational transients and for dry storage. The need is to understand and avoid oxide spallation.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:
This project will take advantage of high-burnup cladding samples that have already been sent to hot cell laboratories as part of various high-burnup post irradiation examination projects (thus saving the cost of fuel extraction and shipment).
Irradiated Zircaloy-4 as well as advanced cladding samples (Zirlo and M-5) will be subjected to additional heat flux and coolant temperature and chemistry conditions representative of a high-duty PWR operating environment under laboratory conditions. Advanced cladding materials may be subjected to more severe environments to investigate the conditions under which spallation in such materials may occur. The samples will be induced to spall and the spallation behavior of the advanced claddings will be evaluated and compared to that of Zircaloy-4.

<u>Work Scope for Subsequent Periods:</u>
The in-laboratory corrosion / spallation experiments will be continued in order to identify the major factors (such as heat flux, cladding temperature, coolant chemistry, extent of localized boiling, etc.) that may be contributing to oxide spallation.

FY02 Deliverable:
A report on spallation behavior in advanced claddings.

FY03 Deliverable:
A report on factors leading to spallation.

FY02 Estimated Cost:	DOE: $500K	EPRI: $500K
Total Estimated Cost:	DOE: $1,000K	EPRI: $1,000K

5-234 Feasibility Determination For the Use of Enriched Boric Acid to Avoid PWR Axial Offset Anomaly (AOA)

Principle Objective: The objective of this study is to determine whether the addition of Enriched Boric Acid to the reactor coolant of commercial PWRs will reduce or eliminate the risk of axial offset anomaly (AOA).

Need: AOA has become a limiting operational condition preventing PWRs from operating with efficient core designs. The economic impact is significant and can weigh on the order of $500K to $2M per operating cycle. Due to it's impact, finding solutions to avoid AOA are a high industry priority. Since 1998, utilities have appropriated approximately $7M toward understanding and modeling the phenomenon. The use of Enriched Boric Acid is considered a viable candidate to avoid AOA. However, due to the high initial capital investment for a PWR to convert to Enriched Boric Acid, in-reactor "proof" testing at a research facility is necessary before implementation in a high-duty commercial core.

Expected Duration: 2.5 years.

Note: Phase I of this project will commence in 2002 and conclude approximately 12-18 months later.
Phase II will be initiated provided successful results are obtained during Phase I.

Scope of Work:

<u>FY2002 Scope of Work</u>:

[In 2001, through the Robust Fuel Program initiative at EPRI, the nuclear industry is funding $2.1M to understand the root causes of AOA and to investigate measures to prevent it. In support of this proposed in-reactor project, EPRI is funding an Enriched Boric Acid test design and specifications project.]

Phase I of this NEPO Project is a feasibility demonstration that will examine the in-reactor facility's ability to imitate the conditions (and symptoms) that cause AOA in a commercial reactor. These conditions are comprised of both chemistry and nuclear parameters. The success of Phase I will be measured by in-reactor monitoring equipment, as well as by post-irradiation fuel examinations**.** Phase I will consist of the following:

1. The in-reactor loop will consist of materials and water chemistry similar to a commercial PWR. In terms of chemistry, Phase I will utilize "normal" boric acid used in commercial reactors as opposed to Enriched Boric Acid.

2. The in-reactor irradiated fuel bundle will generate heat fluxes and sub-cooled nucleate boiling similar to that expected from a high-duty commercial reactor.

3. Neutron detectors, thermal-couples, and/or other devices sensitive enough to measure minor neutron flux depressions will be compared to expected values.

Phase I will be successful if the in-core monitoring equipment indicate the reactor behavior is similar to that of a commercial reactor exhibiting AOA, and post-irradiation examination of the fuel deposits are consistent with those observed from commercial reactors. Satisfactory demonstration of Phase I will prompt a funding submittal to advance to Phase II.

Work Scope for Subsequent Periods:

Phase II will demonstrate the promise of Enriched Boric Acid to prevent or minimize the susceptibility of PWRs to AOA. In-reactor test conditions will be similar to those in Phase I, but the chemistry conditions will change with the replacement of normal boric acid with Enriched Boric Acid.

FY02 Deliverable:

A report on the completed test rig design and construction will be submitted in December 2002. Commencement of Phase I testing is expected in 2002 with completion by mid-year 2003. A Phase I report will be submitted detailing the results at that time.

FY02 Estimated Cost: $341K

Total Estimated Cost: $1,091K

5-235 Dissolution of Fuel Cladding Oxide in High Duty PWR Cores

Principle Objective: To determine the solubility of zirconium dioxide under high duty reactor assembly conditions and the degree to which zirconium dioxide adsorbs boric acid at these conditions.

Need: A stunning finding of the EPRI Robust Fuel Program's 1999-2000 efforts to understand the mechanism of Axial Offset Anomaly (AOA) was that fuel deposits on high duty assemblies implicated in AOA were laced with a fine precipitate of zirconium dioxide (ZrO2) in addition to nickel oxide. These deposits have now been observed from two PWRs operating with high energy cores. The logical source is local dissolution of the fuel cladding oxide, which intuitively seems unlikely. However, there are in fact no reliable thermodynamic data on ZrO2 solubility under near-clad conditions of pH and temperature in high steaming assemblies. The main objective of this project is to make use of the unique capabilities of the high temperature thermodynamics group at the National Labs to produce these missing data. In addition, it is known that ZrO2 has a high affinity at room temperature for adsorption of non-ionic boric acid (Boron is the PWR neutron moderator). If this affinity for boron adsorption occurs at high temperature as well (note that ZrO2 adsorbs lithium at high temperature), then the ZrO2 precipitates found in AOA fuel deposits might be a substrate for the anomalous moderator hideout producing AOA. The National Labs are also well situated to make specific adsorption measurements of boric acid on zirconia at high temperatures.

Axial Offset Anomaly (AOA) in high duty PWR fuel, which threatens shutdown margin during the operating cycle and has forced one PWR to derate, is one of the most vexing problems in fuel performance today. It is a limiting condition preventing many PWRs from operating with more efficient core designs. The economic impact is significant and weighs on the order of $500K to $2M per operating cycle. This project is aimed at understanding the environmental conditions which leads to fuel cladding oxide dissolution and the potential role it plays in contributing to AOA. Ultimately, the results from this work may permit the industry to systematically alter the coolant chemistry to prevent dissolution and AOA.

Expected Duration: 1.5 years

Scope of Work:

FY2002 Scope of Work:
Task 1: Produce chemical thermodynamic solubility data on well-characterized zirconium dioxide at near-clad temperatures and chemistry environment. Produce a model for the temperature dependence of such data above 200 C using National Lab's unique high temperature thermodynamic facilities. Relate results to previous EPRI data on zirconia loading of AOA fuel deposits.

Task 2: Produce specific adsorption isotherms for equilibrium uptake of non-ionic boric acid by well characterized zirconium dioxide powder up to 350 C using ORNLs gold bag autoclave. Relate results to previous EPRI data on zirconia loading of AOA fuel deposits.

FY02 Deliverables:
Technical report(s) describing the results of the two tasks and the implication of those results on the mechanism and prevention of AOA.

FY02 Estimated Cost:	DOE: $115K	EPRI $115K
Total Estimated Cost:	DOE: $175K	EPRI $175K

5-236 Optimizing Coolant Chemistry in BWRs Using Depleted Zinc Oxide and Noble Metal Chemical Application (NMCA)

Principal Objective: Provide guidance for optimizing depleted zinc oxide (DZO) injection levels in BWRs that have injected NMCA to minimize shutdown dose rates in the short term and significantly reduce the amount of costly DZO injected by BWRs in the long term while continuing to meet ALARA goals.

Need: Utilities typically spend ~$250K per fuel cycle for DZO to lower shutdown dose rates, thereby minimizing collective exposures. The proposed project will gather information to confirm the synergistic effects of NMCA and DZO in reducing shutdown dose rates. Analyses of data are expected to lead to reductions in the amount of DZO to be injected while still meeting ALARA goals. Lower injection levels of DZO will significantly reduce the operating costs for BWRs.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:

The scope of work of this activity is to gather and analyze coolant chemistry and shutdown dose rate data from a number of BWRs to confirm the synergistic effects of DZO and NMCA on reduced shutdown dose rates. Guidance would be provided regarding the optimum DZO injection levels.

<u>Work Scope for 2003</u>:

Analyze data and document the results of the study.

FY03 Deliverable:
Report documenting the results of this project.

FY02 Estimated Cost: DOE: $200 K EPRI: $50 K

FY03 Estimated Cost: DOE: $50 K EPRI: $50 K

5-237 Root Cause Investigation of Fuel Rod Oxide Spallation and Noble Metal Chemical Application (NMCA)

Principal Objective: Perform hot cell post-irradiation examination (PIE) of high burnup fuel rods from Peach Bottom-2 (PB2)

Need: Fuel rods from PB2 exhibited oxide spallation after one cycle of operation following treatment of the core internals with noble metal chemical application (NMCA). By end of 2001, 23 of all 34 BWRs will have NMCA to protect core internals from SCC. Also, plants are making effort to extend fuel burnup to gain cycle economics. Combination of NMCA and fuel burnup extension may impact fuel reliability, and evidence of oxide spallation may be an early indication. Results from this PIE will help resolve the root cause of the oxide spallation and result in improvements in the long-term effectiveness of NMCA.

Expected Duration: 2 years

Scope of Work:

<u>FY2002 Scope of Work</u>:
The scope of work of this activity is:

- Retrieve and ship two fuel rods to a hot cell facility for examination
- Perform detailed visual inspection
- Cut and machine samples for detailed examination and analyses

<u>Work Scope for Subsequent Periods:</u>

<u>FY 2003</u>
- Conduct metallography and scanning electron microscopy (SEM) to determine the structure of spalled oxide and crud deposits and the noble metals in the deposits, and hydrogen analysis to determine the extent of hydriding of the cladding. Analyze the role of noble metals, crud and hydriding on oxide spallation and effect on fuel reliability
- Disposal of fuel samples and other radioactive wastes

FY2002 Deliverable:

Hot cell visual inspection results and sample cutting plan.

FY2003 Deliverable:

Report documenting results of the project.

FY02 Estimated Cost: DOE: $500K EPRI: $500K

Total Estimated Cost: DOE: $1,000K EPRI: $1,000K